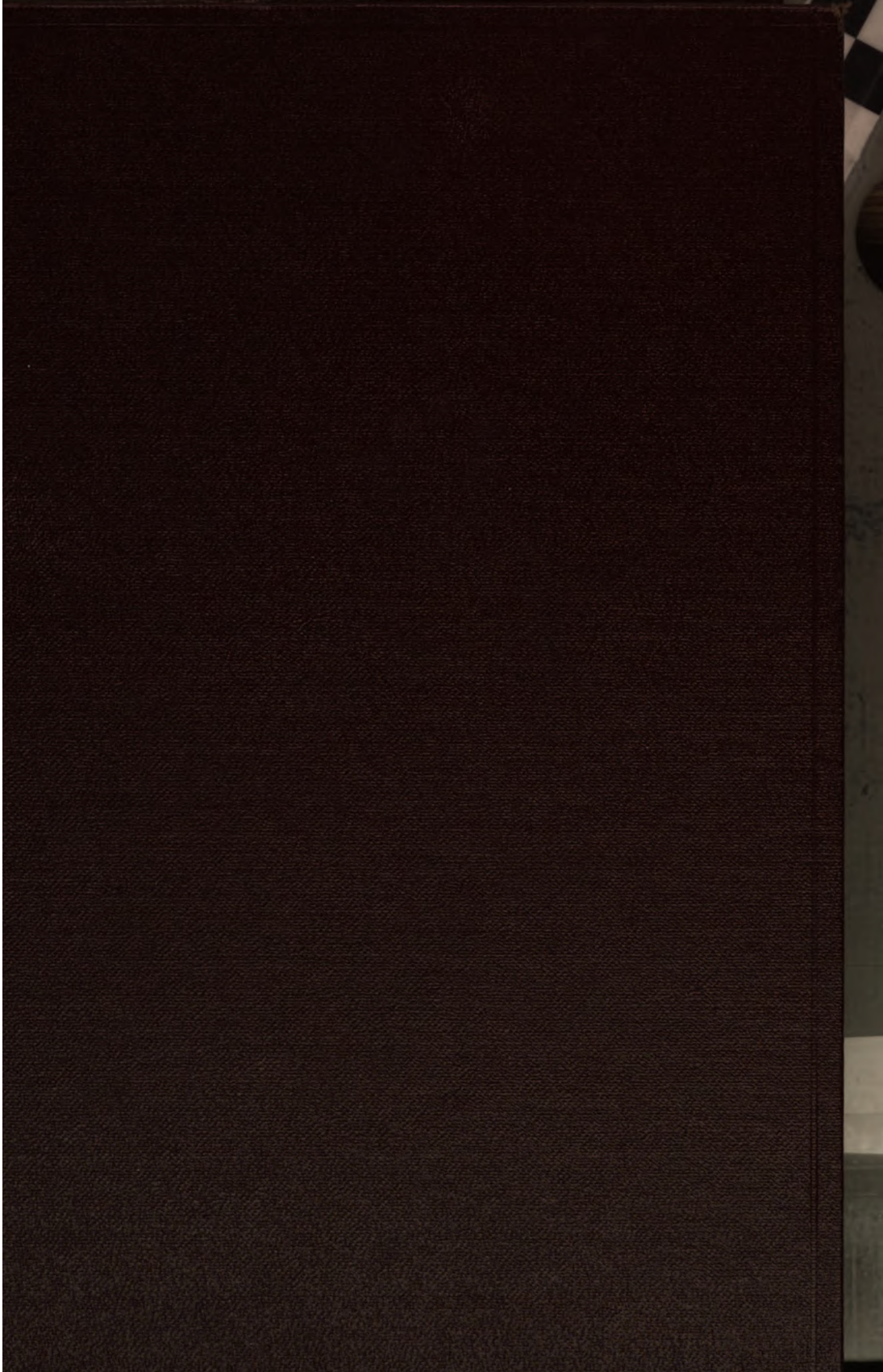
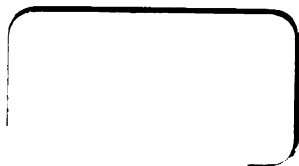






PUBLIC HEALTH LIBRARY



RADIATION HYGIENE HANDBOOK

OTHER MCGRAW-HILL HANDBOOKS OF INTEREST

ABBOTT AND STETKA · National Electrical Code Handbook
AMERICAN INSTITUTE OF PHYSICS · American Institute of Physics Handbook
BEEMAN · Industrial Power Systems Handbook
BLATZ · Radiation Hygiene Handbook
BRADY · Materials Handbook
CARROLL · Industrial Instrument Servicing Handbook
COCKRELL · Industrial Electronics Handbook
CONDON AND ODISHAW · Handbook of Physics
CONSIGNED · Process Instruments and Controls Handbook
CONSIGNED AND ROSS · Handbook of Applied Instrumentation
CROFT AND CARR · American Electrician's Handbook
ETHERINGTON · Nuclear Engineering Handbook
FACTORY MUTUAL ENGINEERING DIVISION · Handbook of Industrial Loss Prevention
FINK · Television Engineering Handbook
HARRIS · Handbook of Noise Control
HARRIS AND CREDE · Shock and Vibration Handbook
HENNEY · Radio Engineering Handbook
HENNEY AND WALSH · Electronic Components Handbook
HUNTER · Handbook of Semiconductor Electronics
HUSKEY AND KORN · Computer Handbook
JASIK · Antenna Engineering Handbook
JURAN · Quality Control Handbook
KALLEN · Handbook of Instrumentation and Controls
KNOWLTON · Standard Handbook for Electrical Engineers
KOELLE · Handbook of Astronautical Engineering
KORN AND KORN · Mathematical Handbook for Scientists and Engineers
KURTZ · The Lineman's Handbook
LANDEE, DAVIS, AND ALBRECHT · Electronic Designers' Handbook
LASSER · Business Management Handbook
MANTELL · Engineering Materials Handbook
MARKS AND BAUMEISTER · Mechanical Engineers' Handbook
MARKUS · Handbook of Electronic Control Circuits
MARKUS AND ZELUFF · Handbook of Industrial Electronic Circuits
MARKUS AND ZELUFF · Handbook of Industrial Electronic Control Circuits
MAYNARD · Industrial Engineering Handbook
MAYNARD · Top Management Handbook
MORROW · Maintenance Engineering Handbook
PERRY · Engineering Manual
ROTHBART · Mechanical Design and Systems Handbook
STANIAR · Plant Engineering Handbook
TERMAN · Radio Engineers' Handbook
TRUXAL · Control Engineers' Handbook

RADIATION HYGIENE HANDBOOK

A PRACTICAL REFERENCE COVERING THE INDUSTRIAL, MEDICAL, AND
RESEARCH USES OF RADIATION AND ATOMIC ENERGY WITH SPECIAL
APPLICATIONS TO THE FIELDS OF HEALTH PHYSICS, INDUSTRIAL
HYGIENE, AND SANITARY ENGINEERING

HANSON BLATZ, Editor-in-Chief

*Director, New York City Office of Radiation Control
Associate Professor of Industrial Medicine, N.Y.U. College of Medicine
Formerly Chief, Radiation Branch, AEC Health and Safety Laboratory*

FIRST EDITION

McGRAW-HILL BOOK COMPANY

NEW YORK SAN FRANCISCO TORONTO LONDON

ref. acc. no. 942

PUBLIC HEALTH

RADIATION HYGIENE HANDBOOK

Copyright © 1959 by McGraw-Hill, Inc. All Rights Reserved. Printed in the United States of America. This book, or parts thereof, may not be reproduced in any form without permission of the publishers.

Library of Congress Catalog Card Number: 59-9985

II
05885

17231
12.125
Public
for the
Library

CONTRIBUTORS

- Harry N. Bane, Memorial Center, New York, N. Y.
- Edgar C. Barnes, Westinghouse Electric Corporation, Pittsburgh,
Pennsylvania.
- Hanson Blatz, New York City Office of Radiation Control, New York, N. Y.
- E. P. Blizard, Oak Ridge National Laboratory, Oak Ridge, Tennessee.
- William M. Breazeale, The Babcock and Wilcox Company, Lynchburg,
Virginia.
- Dixon Callihan, Oak Ridge National Laboratory, Oak Ridge, Tennessee.
- Frederick P. Cowan, Brookhaven National Laboratory, Upton, New York.
- George M. Corney, Eastman Kodak Company, Rochester, New York.
- Rolf Eliassen, Massachusetts Institute of Technology, Cambridge,
Massachusetts.
- Naomi A. Hallden, AEC Health and Safety Laboratory, New York, N. Y.
- David Halliday, University of Pittsburgh, Pittsburgh, Pennsylvania.
- Saul J. Harris, Atomic Industrial Forum, New York, N. Y.
- Myron B Hawkins, University of California, Richmond, California.
- Truman P. Kohman, Carnegie Institute of Technology, Pittsburgh,
Pennsylvania.
- Simon Kinsman, U.S. Public Health Service, San Francisco, California.
- Serge A. Korff, New York University, New York, N. Y.
- Jack S. Krohmer, University of Texas, Dallas, Texas.
- Robert A. Lauderdale, University of Kentucky, Lexington, Kentucky.
- M. Stanley Livingston, Massachusetts Institute of Technology, Cambridge,
Massachusetts.
- Morris F. Milligan, Los Alamos Scientific Laboratory, Los Alamos, New
Mexico.
- G. W. Morgan, Atomic Energy Commission, Washington, D. C.

v

Karl Z. Morgan, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

George A. Morton, R.C.A. Laboratories, Princeton, New Jersey.

James J. Nickson, M.D., Memorial Center, New York, N.Y.

John Ozeroff, General Electric Company, Pleasanton, California.

Hugh C. Paxton, Los Alamos Scientific Laboratory, Los Alamos, New Mexico.

Thomas H. Rogers, Machlett Laboratories, Springdale, Connecticut.

Harald H. Rossi, Columbia University, New York, N.Y.

James H. Schulman, U.S. Naval Research Laboratory, Washington, D.C.

Harry F. Schulte, Los Alamos Scientific Laboratory, Los Alamos, New Mexico.

C. L. Schuske, Dow Chemical Company, Rocky Flats, Colorado.

Sanford C. Sigoloff, Edgerton, Germeshausen and Grier, Inc., Las Vegas, Nevada.

Leslie Silverman, Harvard School of Public Health, Boston, Massachusetts.

James H. Sterner, M.D., Eastman Kodak Company, Rochester, New York.

William L. Sutton, M.D., Eastman Kodak Company, Rochester, New York.

Lauriston S. Taylor, National Bureau of Standards, Washington, D.C.

Paul C. Tompkins, U.S. Naval Radiological Defense Laboratory, San Francisco, California.

Harold O. Wyckoff, National Bureau of Standards, Washington, D.C.

FOREWORD

The scope and complexity of radiation hygiene make it a fascinating field of study for those who enjoy the challenge of scientific variety.

Here is a subject that begins with the properties of the fundamental particles and the way in which they interact with matter—especially the matter of which living things are composed. Although these interactions take place at the lowest biochemical level, they may express themselves so as to involve the entire organism and, in the case of effects on the hereditary mechanisms, the entire race.

It is in the study of the dispersion of radioactive substances in the environment that one finds the greatest degree of interplay among the many branches of science. Meteorology, soil chemistry, the hydrologic sciences, and mineral metabolism are some of the subjects which are involved to a major degree in evaluating the significance of radioactive contamination.

The ultimate objective of the radiation hygienist is to recommend effective and yet economical methods of controlling radiation hazards. This frequently requires intimate knowledge of the procedures of laboratories and industrial processes. The job of the radiation hygienist does not stop with identification of a potential hazard: it is important that he assist as well in the solution of the problem.

Radiation hygienists today are for the most part specialists of one kind or another who have acquired more general knowledge through their working relationships with specialists from other fields. Logically, the first of the workers in the field of radiation protection were physicists, many of whom were concerned originally with dosimetric aspects of radiation exposure. In recent years the ranks of this new profession have attracted chemists, geologists, engineers, meteorologists, and biologists.

The interdisciplinary characteristic of the field creates problems of definition and purpose which must soon be solved if the needs of the expanding nuclear industry are to be met. Who is a radiation hygienist (or, for that matter, what is a health physicist)? What should he know, what should he do, where do his responsibilities begin, and where do they end? As our second and third generations of scientists and engineers enter this expanding field, our definitions must be improved, our standards for pro-

professional qualification must be established, and our universities must adapt their curricula for the new requirements that the atomic age imposes.

The "Radiation Hygiene Handbook" will serve in an important way to assist science and industry to find the basic information with which radiation protection problems can be solved. For the first time, a compendium of information is available to permit the scientist and engineer to probe the many ramifications of health protection in the peaceful applications of atomic energy.

Merril Eisenbud

PREFACE

This is a new handbook to fill a new need. With the rapidly increasing use of radiation as a tool in industry, medicine, and research and the frequent appearance of radiation as an objectionable by-product in such fields as atomic energy for power and for transportation, the need for information and data to help protect workers and the public from its harmful effects is pressing.

This, the first comprehensive handbook in the field we chose to call "radiation hygiene," has been dependent upon the contributions, advice, and inspiration of many of the American leaders in this relatively new branch of science. The list of contributors, their areas of specialization, and their affiliations give some indication of the broadness of the subject.

Many different disciplines and many different philosophies are represented in this new field. On the one hand are the ultraconservatives, who believe that because of the many unknowns (particularly with regard to genetics) radioactive contamination and radiation exposures should be kept to the absolute minimum, regardless of cost. At the other extreme are those who prefer to wait for evidence of damage before instituting corrective measures.

In offering guidance to the various contributors, the editor has attempted to follow a reasonably "middle-of-the-road" philosophy as exemplified by many of the leaders in this field with which the editor has had the good fortune to be associated for many years: Shields Warren, John Bugher, G. Failla, Charles L. Dunham, Carl B. Braestrup, S. Allan Lough, William B. Harris, John H. Harley, Norton Nelson, and many others.

If any particular point of view is expressed in this handbook, it was perhaps influenced most by Merrill Eisenbud, through whose leadership the AEC Health and Safety Laboratory has to a great extent set the pace in this rapidly expanding field of endeavor. To him, in particular, this book is dedicated.

Many of my former staff in the Radiation Branch of the Health and Safety Laboratory have been most generous in their advice, encouragement, and assistance, particularly Leonard R. Solon, James E. McLaughlin, Yvette Rosenberg, Pauline Castellani, and Josephine Lemma. Last, but by no means least, this formidable task could not have been completed without the infinite and loving patience of my wife Elizabeth and our family through these trying three years.

Hanson Blatz

CONTENTS

<i>Contributors</i>	v
<i>Foreword by Merrill Eisenbud</i>	vii
<i>Preface</i>	ix
Section 1. <i>Reference Data</i>	1-1
Section 2. <i>Glossary of Terms</i>	2-1
Section 3. <i>Exposure Standards and Radiation Protection Regulations</i>	3-1
Section 4. <i>Natural Radioactive Background</i>	4-1
Section 5. <i>Ionizing Radiation</i>	5-1
Section 6. <i>Sources of Radiation</i>	6-1
Section 7. <i>Interaction of Radiation with Matter</i>	7-1
Section 8. <i>Radiation Attenuation Data</i>	8-1
Section 9. <i>Laboratory Design</i>	9-1
Section 10. <i>Radiation Detection and Measurements</i>	10-1
Section 11. <i>Industrial Applications</i>	11-1
Section 12. <i>Research Applications</i>	12-1
Section 13. <i>Medical Radiation Applications</i>	13-1
Section 14. <i>Determination of Exposures</i>	14-1
Section 15. <i>Nuclear Safety</i>	15-1
Section 16. <i>Radiation Hygiene Chemistry</i>	16-1
Section 17. <i>Equipment for Handling, Storage, and Transportation of Radioactive Materials</i>	17-1
Section 18. <i>Surface Contamination and Decontamination</i>	18-1
Section 19. <i>Physiological Effects of Radiation</i>	19-1
Section 20. <i>Sampling Equipment (Dust, Gases, and Liquids)</i>	20-1
Section 21. <i>Liquid and Solid Waste Disposal</i>	21-1
Section 22. <i>Control of Radioactive Air Pollution</i>	22-1
Section 23. <i>Personnel Control</i>	23-1

Index follows Section 23.

RADIATION HYGIENE HANDBOOK

1

1.1
1.2
1.3

1.4
1.5
1.6
1.7
1.8

1.9
1.10
1.11
1.12

1.13
1.14

1.15
1.16
1.17
1.18

Section 1

REFERENCE DATA

By

HANSON BLATZ, *Director, New York City Office of Radiation Control, New York.*

SIMON KINSMAN, Ph.D., *Scientist, United States Public Health Service, Region IX,
San Francisco, California.*

Table 1-1.	Alphabetical Index of Elements and Their Atomic Numbers.....	1-3
Table 1-2.	Half-lives of Common Radioactive Elements.....	1-4
Table 1-3.	Fundamental Constants.....	1-5
Table 1-4.	Signs and Symbols of Particular Interest in Radiation and Radioactivity.....	1-6
Table 1-5.	Commonly Used Units (Radioactivity, Radiation, and Energy).....	1-7
Table 1-6.	Logarithm Tables (Four-place).....	1-8
Table 1-7.	Natural Trigonometric Functions.....	1-10
Table 1-8.	Numbers, Squares, Square Roots, Cubes, and Cube Roots.....	1-12
Table 1-9.	Values and Logarithms of Exponential Functions.....	1-25
Table 1-10.	Air-density Correction Factors for Air-ionization Measurements.....	1-29
Table 1-11.	Properties of the Earth's Atmosphere.....	1-30
Table 1-12.	Weather Data—U.S. Weather Bureau....	1-31
Table 1-13.	Densities of Common Metals.....	1-35
Table 1-14.	Sheet-lead Thicknesses.....	1-36
Table 1-15.	Relation between Thickness of Ordinary Concrete and Thickness of Lead for Radium and Cobalt-60 Gamma Rays (Broad-beam Conditions).....	1-36

Table 1-16.	Relation between Thickness of Ordinary Concrete and Lead Equivalent at Various X-ray Qualities (Broad-beam Conditions)	1-37
Table 1-17.	Relation between Thickness of Brick and Lead Equivalent at Various X-ray Qualities (Broad-beam Conditions)	1-37
Table 1-18.	The Standard Man: Masses of Organs of Human Body	1-38
Table 1-19.	The Standard Man: Daily Rates of Ingestion and Inhalation	1-39
Table 1-20.	The Standard Man: Particulates in Respiratory Tract	1-39
Table 1-21.	Count-rate Nomographs	1-40
Table 1-22.	Multiple Half-lives and Half-value Layers	1-42

Acknowledgments. Some of the material in this section is from the "Radiological Health Handbook" of the U.S. Department of Health, Education and Welfare. Dr. Kinsman, as editor of the "Radiological Health Handbook," wishes to acknowledge the assistance of his many associates at the Robert A. Taft Sanitary Engineering Center in Cincinnati, Ohio.

Tables 1-15, 1-16, and 1-17 are based, in part, on data from the "Code of Practice for the Protection of Persons Exposed to Ionizing Radiations," British Radioactive Substances Standing Advisory Committee, H.M. Stationery Office, London, England (1957).

Tables 1-18, 1-19, and 1-20 are from the recommendations of the International Commission on Radiological Protection.

REFERENCE DATA

Hanson Blatz and Simon Kinsman

Table 1-1. Alphabetical Index of Elements and Their Atomic Numbers (Z)

Element	Symbol	Z	Element	Symbol	Z
Actinium.....	Ac	89	Molybdenum.....	Mo	42
Aluminum.....	Al	13	Neodymium.....	Nd	60
Americium.....	Am	95	Neon.....	Ne	10
Antimony.....	Sb	51	Neptunium.....	Np	93
Argon.....	A	18	Nickel.....	Ni	28
Arsenic.....	As	33	Niobium *.....	Nb	41
Astatine.....	At	85	Nitrogen.....	N	7
Barium.....	Ba	56	Osmium.....	Os	76
Berkelium.....	Bk	97	Oxygen.....	O	8
Beryllium.....	Be	4	Palladium.....	Pd	46
Bismuth.....	Bi	83	Phosphorus.....	P	15
Boron.....	B	5	Platinum.....	Pt	78
Bromine.....	Br	35	Plutonium.....	Pu	94
Cadmium.....	Cd	48	Polonium.....	Po	84
Calcium.....	Ca	20	Potassium.....	K	19
Californium.....	Cf	98	Praseodymium....	Pr	59
Carbon.....	C	6	Promethium.....	Pm	61
Cerium.....	Ce	58	Protoactinium....	Pa	91
Cesium.....	Cs	55	Radium.....	Ra	88
Chlorine.....	Cl	17	Radon †.....	Rn	86
Chromium.....	Cr	24	Rhenium.....	Re	75
Cobalt.....	Co	27	Rhodium.....	Rh	45
Copper.....	Cu	29	Rubidium.....	Rb	37
Curium.....	Cm	96	Ruthenium.....	Ru	44
Dysprosium.....	Dy	66	Samarium.....	Sm	62
Erbium.....	Er	68	Scandium.....	Sc	21
Europium.....	Eu	63	Selenium.....	Se	34
Fluorine.....	F	9	Silicon.....	Si	14
Francium.....	Fr	87	Silver.....	Ag	47
Gadolinium.....	Gd	64	Sodium.....	Na	11
Gallium.....	Ga	31	Strontium.....	Sr	38
Germanium.....	Ge	32	Sulfur.....	S	16
Gold.....	Au	79	Tantalum.....	Ta	73
Hafnium.....	Hf	72	Technetium.....	Tc	43
Helium.....	He	2	Tellurium.....	Te	52
Holmium.....	Ho	67	Terbium.....	Tb	65
Hydrogen.....	H	1	Thallium.....	Tl	81
Indium.....	In	49	Thorium.....	Th	90
Iodine.....	I	53	Thulium.....	Tm	69
Iridium.....	Ir	77	Tin.....	Sn	50
Iron.....	Fe	26	Titanium.....	Ti	22
Krypton.....	Kr	36	Uranium.....	U	92
Lanthanum.....	La	57	Vanadium.....	V	23
Lead.....	Pb	82	Wolfram ‡.....	W	74
Lithium.....	Li	3	Xenon.....	Xe	54
Lutecium.....	Lu	71	Ytterbium.....	Yb	70
Magnesium.....	Mg	12	Yttrium.....	Y	39
Manganese.....	Mn	25	Zinc.....	Zn	30
Mercury.....	Hg	80	Zirconium.....	Zr	40

* Formerly columbium (Cb). † Also called radium emanation. ‡ Also called tungsten.

Table 1-2. Half-lives of Common Radioactive Elements

Element	Half-life	Element	Half-life
Th-232	1.39×10^{10} y	Y-91	58 d
U-238	4.49×10^9 y	Sr-89	53 d
U-235	7.13×10^8 y	Fe-59	45 d
Cl-36	3.1×10^5 y	Cr-51	27 d
U-233	1.62×10^5 y	Th-234	24.1 d
Ni-59	8×10^4 y	Rb-86	18.6 d
Pu-239	2.436×10^4 y	P-32	14.5 d
C-14	5,600 y	Ba-140	12.8 d
Ra-226	1,622 y	I-131	8.05 d
Cs-137	30 y	Au-199	3.15 d
Sr-90	28 y	Au-198	2.7 d
H-3	12.26 y	Mo-99	67 h
Co-60	5.27 y	Y-90	64 h
Tl-204	4.1 y	La-140	40 h
Fe-55	2.9 y	Br-82	35.87 h
Pm-147	2.6 y	As-76	26.8 h
Cs-134	2.3 y	Na-24	15.0 h
Ru-106	1.0 y	Ga-72	14.1 h
Ce-144	285 d	Cu-64	12.8 h
Zn-65	245 d	K-42	12.5 h
Ca-45	160 d	Mn-56	2.576 h
Po-210	138.4 d	A-41	109 m
Ta-182	112 d	Pr-144	17 m
S-35	87 d	N-16	7.4 s
W-185	74 d	Po-212	3×10^{-7} s

Table 1-3. Fundamental Constants

Name	Value
Avogadro's number.....	$N_0 = 6.025 \times 10^{23}$ molecules/g mole
Base of natural logarithms.....	$e = 2.7183...$
Curie.....	$c = 3.7 \times 10^{10}$ disintegrations/sec
Electron charge.....	$e = 4.8 \times 10^{-10}$ statcoulomb $= 1.6 \times 10^{-19}$ coulomb
Energy equivalent of electron mass	$mc^2 = 0.51$ Mev
Faraday's constant.....	$F = 96,514$ coulombs/g equivalent (physical scale)
Frequency associated with 1 ev...	$\nu_0 = 2.4186 \times 10^{14}$ sec $^{-1}$
Gravitational acceleration.....	$g = 980.665$ cm/sec 2
Mass, alpha particle.....	$m_\alpha = 6.64 \times 10^{-24}$ g = 4.002777 mu
Mass, electron.....	$m_e = 9.1066 \times 10^{-28}$ g = 0.000548 mu
Mass, H-atom.....	$m_H = 1.67339 \times 10^{-24}$ g = 1.008142 mu
Mass, neutron.....	$m_n = 1.6751 \times 10^{-24}$ g = 1.008982 mu
Mass, proton.....	$m_p = 1.67248 \times 10^{-24}$ g = 1.007594 mu
Mass unit.....	mu = 1.66035 $\times 10^{-24}$ g = 1.0000 mu
Microcurie.....	$\mu c = 10^{-6}$ curie = 3.7×10^4 disintegrations/sec
Micromicrocurie.....	$\mu\mu c = 10^{-12}$ curie = 3.7×10^{-2} disintegration/sec
Millicurie.....	mc = 10^{-3} curie = 3.7×10^7 disintegrations/sec
Planck's constant.....	$h = 6.624 \times 10^{-27}$ erg-sec
Roentgen.....	r = 1 esu/0.001293 g of air
Rutherford.....	rd = 10^6 disintegrations/sec
Universal gas constant.....	$R = 0.08206$ liter-atm/g-mole/ $^{\circ}$ K
Velocity of light.....	$c = 2.99776 \times 10^{10}$ cm/sec
Wavelength associated with 1 ev..	$\lambda_0 = 12394.8$ A

Table 1-4. Signs and Symbols of Particular Interest in Radiation and Radioactivity

A or $\text{\AA}$	angstrom (10^{-10} m)
A	activity (radioactivity)
A_0	activity (radioactivity), original
A_t	activity (radioactivity), at time t
b	build-up factor
c	curie
D	deuterium
d	deuteron
e	electron
I	radiation intensity
I_0	radiation intensity, initial
I_x	radiation intensity, transmitted
LD-50	median lethal dose
n	neutron
N	number of counts
N_0	number of counts, original
N_t	number of counts, at time t
h	Planck's constant
p	proton
r	roentgen
t	triton
α	alpha particle
β	beta particle
γ	gamma ray
λ	wavelength or decay constant
$\bar{\lambda}$	mean free path
u	linear absorption coefficient
σ	area (barn-cross section)

Table 1-5. Commonly Used Units

Units of radioactivity:

$\mu\mu\text{c}$	micromicrocurie.....	10^{-12} curie
μc	microcurie.....	10^{-6} curie
c	curie	
kc	kilocurie.....	10^3 curies
Mc	megacurie.....	10^6 curies

Units of radiation:

μr	microroentgen.....	10^{-6} roentgen
mr	milliroentgen.....	10^{-3} roentgen
r	roentgen	

The prefixes milli (m) and micro (μ) are also commonly used with the units rad, rep (obsolescent), and rem.

Units of energy (of radiation):

ev	electron volt	
kev	kiloelectron volt.....	10^3 electron volts
Mev	million electron volts.....	10^6 electron volts
Bev	billion electron volts.....	10^9 electron volts

Table 1-6. Logarithm Tables (Four-place) *

N	0	1	2	3	4	5	6	7	8	9
10	0000	0043	0086	0128	0170	0212	0253	0294	0334	0374
11	0414	0453	0492	0531	0569	0607	0645	0682	0719	0755
12	0792	0828	0864	0899	0934	0969	1004	1038	1072	1106
13	1139	1173	1206	1239	1271	1303	1335	1367	1399	1430
14	1461	1492	1523	1553	1584	1614	1644	1673	1703	1732
15	1761	1790	1818	1847	1875	1903	1931	1959	1987	2014
16	2041	2068	2095	2122	2148	2175	2201	2227	2253	2279
17	2304	2330	2355	2380	2405	2430	2455	2480	2504	2529
18	2553	2577	2601	2625	2648	2672	2695	2718	2742	2765
19	2788	2810	2833	2856	2878	2900	2923	2945	2967	2989
20	3010	3032	3054	3075	3096	3118	3139	3160	3181	3201
21	3222	3243	3263	3284	3304	3324	3345	3365	3385	3404
22	3424	3444	3464	3483	3502	3522	3541	3560	3579	3598
23	3617	3636	3655	3674	3692	3711	3729	3747	3766	3784
24	3802	3820	3838	3856	3874	3892	3909	3927	3945	3962
25	3979	3997	4014	4031	4048	4065	4082	4099	4116	4133
26	4150	4166	4183	4200	4216	4232	4249	4265	4281	4298
27	4314	4330	4346	4362	4378	4393	4409	4425	4440	4456
28	4472	4487	4502	4518	4533	4548	4564	4579	4594	4609
29	4624	4639	4654	4669	4683	4698	4713	4728	4742	4757
30	4771	4786	4800	4814	4829	4843	4857	4871	4886	4900
31	4914	4928	4942	4955	4969	4983	4997	5011	5024	5038
32	5051	5065	5079	5092	5105	5119	5132	5145	5159	5172
33	5185	5198	5211	5224	5237	5250	5263	5276	5289	5302
34	5315	5328	5340	5353	5366	5378	5391	5403	5416	5428
35	5441	5453	5465	5478	5490	5502	5514	5527	5539	5551
36	5563	5575	5587	5599	5611	5623	5635	5647	5658	5670
37	5682	5694	5705	5717	5729	5740	5752	5763	5775	5786
38	5798	5809	5821	5832	5843	5855	5866	5877	5888	5899
39	5911	5922	5933	5944	5955	5966	5977	5988	5999	6010
40	6021	6031	6042	6053	6064	6075	6085	6096	6107	6117
41	6128	6138	6149	6160	6170	6180	6191	6201	6212	6222
42	6232	6243	6253	6263	6274	6284	6294	6304	6314	6325
43	6335	6345	6355	6365	6375	6385	6395	6405	6415	6425
44	6435	6444	6454	6464	6474	6484	6493	6503	6513	6522
45	6532	6542	6551	6561	6571	6580	6590	6599	6609	6618
46	6628	6637	6646	6656	6665	6675	6684	6693	6702	6712
47	6721	6730	6739	6749	6758	6767	6776	6785	6794	6803
48	6812	6821	6830	6839	6848	6857	6866	6875	6884	6893
49	6902	6911	6920	6928	6937	6946	6955	6964	6972	6981
50	6990	6998	7007	7016	7024	7033	7042	7050	7059	7067
51	7076	7084	7093	7101	7110	7118	7126	7135	7143	7152
52	7160	7168	7177	7185	7193	7202	7210	7218	7226	7235
53	7243	7251	7259	7267	7275	7284	7292	7300	7308	7316
54	7324	7332	7340	7348	7356	7364	7372	7380	7388	7396

* From Knowlton, A. E. (ed.), "Standard Handbook for Electrical Engineers," 9th ed., McGraw-Hill, 1957.

Table 1-6. Logarithm Tables (Four-place) (Continued)

N	0	1	2	3	4	5	6	7	8	9
55	7404	7412	7419	7427	7435	7443	7451	7459	7466	7474
56	7482	7490	7497	7505	7513	7520	7528	7536	7543	7551
57	7559	7566	7574	7582	7589	7597	7604	7612	7619	7627
58	7634	7642	7649	7657	7664	7672	7679	7686	7694	7701
59	7709	7716	7723	7731	7738	7745	7752	7760	7767	7774
60	7782	7789	7796	7803	7810	7818	7825	7832	7839	7846
61	7853	7860	7868	7875	7882	7889	7896	7903	7910	7917
62	7924	7931	7938	7945	7952	7959	7966	7973	7980	7987
63	7993	8000	8007	8014	8021	8028	8035	8041	8048	8055
64	8062	8069	8075	8082	8089	8096	8102	8109	8116	8122
65	8129	8136	8142	8149	8156	8162	8169	8176	8182	8189
66	8195	8202	8209	8215	8222	8228	8235	8241	8248	8254
67	8261	8267	8274	8280	8287	8293	8299	8306	8312	8319
68	8325	8331	8338	8344	8351	8357	8363	8370	8376	8382
69	8388	8395	8401	8407	8414	8420	8426	8432	8439	8445
70	8451	8457	8463	8470	8476	8482	8488	8494	8500	8506
71	8513	8519	8525	8531	8537	8543	8549	8555	8561	8567
72	8573	8579	8585	8591	8597	8603	8609	8615	8621	8627
73	8633	8639	8645	8651	8657	8663	8669	8675	8681	8686
74	8692	8698	8704	8710	8716	8722	8727	8733	8739	8745
75	8751	8756	8762	8768	8774	8779	8785	8791	8797	8802
76	8808	8814	8820	8825	8831	8837	8842	8848	8854	8859
77	8865	8871	8876	8882	8887	8893	8899	8904	8910	8915
78	8921	8927	8932	8938	8943	8949	8954	8960	8965	8971
79	8976	8982	8987	8993	8998	9004	9009	9015	9020	9025
80	9031	9036	9042	9047	9053	9058	9063	9069	9074	9079
81	9085	9090	9096	9101	9106	9112	9117	9122	9128	9133
82	9138	9143	9149	9154	9159	9165	9170	9175	9180	9186
83	9191	9196	9201	9206	9212	9217	9222	9227	9232	9238
84	9243	9248	9253	9258	9263	9269	9274	9279	9284	9289
85	9294	9299	9304	9309	9315	9320	9325	9330	9335	9340
86	9345	9350	9355	9360	9365	9370	9375	9380	9385	9390
87	9395	9400	9405	9410	9415	9420	9425	9430	9435	9440
88	9445	9450	9455	9460	9465	9469	9474	9479	9484	9489
89	9494	9499	9504	9509	9513	9518	9523	9528	9533	9538
90	9542	9547	9552	9557	9562	9566	9571	9576	9581	9586
91	9590	9595	9600	9605	9609	9614	9619	9624	9628	9633
92	9638	9643	9647	9652	9657	9661	9666	9671	9675	9680
93	9685	9689	9694	9699	9703	9708	9713	9717	9722	9727
94	9731	9736	9741	9745	9750	9754	9759	9763	9768	9773
95	9777	9782	9786	9791	9795	9800	9805	9809	9814	9818
96	9823	9827	9832	9836	9841	9845	9850	9854	9859	9863
97	9868	9872	9877	9881	9886	9890	9894	9899	9903	9908
98	9912	9917	9921	9926	9930	9934	9939	9943	9948	9952
99	9956	9961	9965	9969	9974	9978	9983	9987	9991	9996

Table 1-7. Natural Trigonometric Functions *

Deg	°0.0	°0.1	°0.2	°0.3	°0.4	°0.5	°0.6	°0.7	°0.8	°0.9	
0	0.0000	0.0017	0.0035	0.0052	0.0070	0.0087	0.0105	0.0122	0.0140	0.0157	89
1	0.0175	0.0192	0.0209	0.0227	0.0244	0.0262	0.0279	0.0297	0.0314	0.0332	88
2	0.0349	0.0366	0.0384	0.0401	0.0419	0.0436	0.0454	0.0471	0.0488	0.0506	87
3	0.0523	0.0541	0.0558	0.0576	0.0593	0.0610	0.0628	0.0645	0.0663	0.0680	86
4	0.0698	0.0715	0.0732	0.0750	0.0767	0.0785	0.0802	0.0819	0.0837	0.0854	85
5	0.0872	0.0889	0.0906	0.0924	0.0941	0.0958	0.0976	0.0993	0.1011	0.1028	84
6	0.1045	0.1063	0.1080	0.1097	0.1115	0.1132	0.1149	0.1167	0.1184	0.1201	83
7	0.1219	0.1236	0.1253	0.1271	0.1288	0.1305	0.1323	0.1340	0.1357	0.1374	82
8	0.1392	0.1409	0.1426	0.1444	0.1461	0.1478	0.1495	0.1513	0.1530	0.1547	81
9	0.1564	0.1582	0.1599	0.1616	0.1633	0.1650	0.1668	0.1685	0.1702	0.1719	80
10	0.1736	0.1754	0.1771	0.1788	0.1805	0.1822	0.1840	0.1857	0.1874	0.1891	79
11	0.1908	0.1925	0.1942	0.1959	0.1977	0.1994	0.2011	0.2028	0.2045	0.2062	78
12	0.2079	0.2096	0.2113	0.2130	0.2147	0.2164	0.2181	0.2198	0.2215	0.2233	77
13	0.2250	0.2267	0.2284	0.2300	0.2317	0.2334	0.2351	0.2368	0.2385	0.2402	76
14	0.2419	0.2436	0.2453	0.2470	0.2487	0.2504	0.2521	0.2538	0.2554	0.2571	75
15	0.2588	0.2605	0.2622	0.2639	0.2656	0.2672	0.2689	0.2706	0.2723	0.2740	74
16	0.2756	0.2773	0.2790	0.2807	0.2823	0.2840	0.2857	0.2874	0.2890	0.2907	73
17	0.2924	0.2940	0.2957	0.2974	0.2990	0.3007	0.3024	0.3040	0.3057	0.3074	72
18	0.3090	0.3107	0.3123	0.3140	0.3156	0.3173	0.3190	0.3206	0.3223	0.3239	71
19	0.3256	0.3272	0.3289	0.3305	0.3322	0.3338	0.3355	0.3371	0.3387	0.3404	70
20	0.3420	0.3437	0.3453	0.3469	0.3486	0.3502	0.3518	0.3535	0.3551	0.3567	69
21	0.3584	0.3600	0.3616	0.3633	0.3649	0.3665	0.3681	0.3697	0.3714	0.3730	68
22	0.3746	0.3762	0.3778	0.3795	0.3811	0.3827	0.3843	0.3859	0.3875	0.3891	67
23	0.3907	0.3923	0.3939	0.3955	0.3971	0.3987	0.4003	0.4019	0.4035	0.4051	66
24	0.4067	0.4083	0.4099	0.4115	0.4131	0.4147	0.4163	0.4179	0.4195	0.4210	65
25	0.4226	0.4242	0.4258	0.4274	0.4289	0.4305	0.4321	0.4337	0.4352	0.4368	64
26	0.4384	0.4399	0.4415	0.4431	0.4446	0.4462	0.4478	0.4493	0.4509	0.4524	63
27	0.4540	0.4555	0.4571	0.4586	0.4602	0.4617	0.4633	0.4648	0.4664	0.4679	62
28	0.4695	0.4710	0.4726	0.4741	0.4756	0.4772	0.4787	0.4802	0.4818	0.4833	61
29	0.4848	0.4863	0.4879	0.4894	0.4909	0.4924	0.4939	0.4955	0.4970	0.4985	60
30	0.5000	0.5015	0.5030	0.5045	0.5060	0.5075	0.5090	0.5105	0.5120	0.5135	59
31	0.5150	0.5165	0.5180	0.5195	0.5210	0.5225	0.5240	0.5255	0.5270	0.5284	58
32	0.5299	0.5314	0.5329	0.5344	0.5358	0.5373	0.5388	0.5402	0.5417	0.5432	57
33	0.5446	0.5461	0.5476	0.5490	0.5505	0.5519	0.5534	0.5548	0.5563	0.5577	56
34	0.5592	0.5606	0.5621	0.5635	0.5650	0.5664	0.5678	0.5693	0.5707	0.5721	55
35	0.5736	0.5750	0.5764	0.5779	0.5793	0.5807	0.5821	0.5835	0.5850	0.5864	54
36	0.5878	0.5892	0.5906	0.5920	0.5934	0.5948	0.5962	0.5976	0.5990	0.6004	53
37	0.6018	0.6032	0.6046	0.6060	0.6074	0.6088	0.6101	0.6115	0.6129	0.6143	52
38	0.6157	0.6170	0.6184	0.6198	0.6211	0.6225	0.6239	0.6252	0.6266	0.6280	51
39	0.6293	0.6307	0.6320	0.6334	0.6347	0.6361	0.6374	0.6388	0.6401	0.6414	50
40	0.6428	0.6441	0.6455	0.6468	0.6481	0.6494	0.6508	0.6521	0.6534	0.6547	49
41	0.6561	0.6574	0.6587	0.6600	0.6613	0.6626	0.6639	0.6652	0.6665	0.6678	48
42	0.6691	0.6704	0.6717	0.6730	0.6743	0.6756	0.6769	0.6782	0.6794	0.6807	47
43	0.6820	0.6833	0.6845	0.6858	0.6871	0.6884	0.6896	0.6909	0.6921	0.6934	46
44	0.6947	0.6959	0.6972	0.6984	0.6997	0.7009	0.7022	0.7034	0.7046	0.7059	45
	1.0	0.9	0.8	0.7	0.6	0.5	0.4	0.3	0.2	0.1	Deg

* From Knowlton, A. E. (ed.), "Standard Handbook for Electrical Engineers," 9th ed., McGraw-Hill, 1957.

Table 1-7. Natural Trigonometric Functions (Continued)

Deg	°0.0	°0.1	°0.2	°0.3	°0.4	°0.5	°0.6	°0.7	°0.8	°0.9	
45	0.7071	0.7083	0.7096	0.7108	0.7120	0.7133	0.7145	0.7157	0.7169	0.7181	44
46	0.7193	0.7206	0.7218	0.7230	0.7242	0.7254	0.7266	0.7278	0.7290	0.7302	43
47	0.7314	0.7235	0.7337	0.7349	0.7361	0.7373	0.7385	0.7396	0.7408	0.7420	42
48	0.7431	0.7443	0.7455	0.7466	0.7478	0.7490	0.7501	0.7513	0.7524	0.7536	41
49	0.7547	0.7559	0.7570	0.7581	0.7593	0.7604	0.7615	0.7627	0.7638	0.7649	40
50	0.7660	0.7672	0.7683	0.7694	0.7705	0.7716	0.7727	0.7738	0.7749	0.7760	39
51	0.7771	0.7782	0.7793	0.7804	0.7815	0.7826	0.7837	0.7848	0.7859	0.7869	38
52	0.7880	0.7891	0.7902	0.7912	0.7923	0.7934	0.7944	0.7955	0.7965	0.7976	37
53	0.7986	0.7997	0.8007	0.8018	0.8028	0.8039	0.8049	0.8059	0.8070	0.8080	36
54	0.8090	0.8100	0.8111	0.8121	0.8131	0.8141	0.8151	0.8161	0.8171	0.8181	35
55	0.8192	0.8202	0.8211	0.8221	0.8231	0.8241	0.8251	0.8261	0.8271	0.8281	34
56	0.8290	0.8300	0.8310	0.8320	0.8329	0.8339	0.8348	0.8358	0.8368	0.8377	33
57	0.8387	0.8396	0.8406	0.8415	0.8425	0.8434	0.8443	0.8453	0.8462	0.8471	32
58	0.8480	0.8490	0.8499	0.8508	0.8517	0.8526	0.8536	0.8545	0.8554	0.8563	31
59	0.8572	0.8581	0.8590	0.8599	0.8607	0.8616	0.8625	0.8634	0.8643	0.8652	30
60	0.8660	0.8669	0.8678	0.8686	0.8695	0.8704	0.8712	0.8721	0.8729	0.8738	29
61	0.8746	0.8755	0.8763	0.8771	0.8780	0.8788	0.8796	0.8805	0.8813	0.8821	28
62	0.8829	0.8838	0.8846	0.8854	0.8862	0.8870	0.8878	0.8886	0.8894	0.8902	27
63	0.8910	0.8918	0.8926	0.8934	0.8942	0.8949	0.8957	0.8965	0.8973	0.8980	26
64	0.8988	0.8996	0.9003	0.9011	0.9018	0.9026	0.9033	0.9041	0.9048	0.9056	25
65	0.9063	0.9070	0.9078	0.9085	0.9092	0.9100	0.9107	0.9114	0.9121	0.9128	24
66	0.9135	0.9143	0.9150	0.9157	0.9164	0.9171	0.9178	0.9184	0.9191	0.9198	23
67	0.9205	0.9212	0.9219	0.9225	0.9232	0.9239	0.9245	0.9252	0.9259	0.9265	22
68	0.9272	0.9278	0.9285	0.9291	0.9298	0.9304	0.9311	0.9317	0.9323	0.9330	21
69	0.9336	0.9342	0.9348	0.9354	0.9361	0.9367	0.9373	0.9379	0.9385	0.9391	20
70	0.9397	0.9403	0.9409	0.9415	0.9421	0.9426	0.9432	0.9438	0.9444	0.9449	19
71	0.9455	0.9461	0.9466	0.9472	0.9478	0.9483	0.9489	0.9494	0.9500	0.9505	18
72	0.9511	0.9516	0.9521	0.9527	0.9532	0.9537	0.9542	0.9548	0.9553	0.9558	17
73	0.9563	0.9568	0.9573	0.9578	0.9583	0.9588	0.9593	0.9598	0.9603	0.9608	16
74	0.9613	0.9617	0.9622	0.9627	0.9632	0.9636	0.9641	0.9646	0.9650	0.9655	15
75	0.9659	0.9664	0.9668	0.9673	0.9677	0.9681	0.9686	0.9690	0.9694	0.9699	14
76	0.9703	0.9707	0.9711	0.9715	0.9720	0.9724	0.9728	0.9732	0.9736	0.9740	13
77	0.9744	0.9748	0.9751	0.9755	0.9759	0.9763	0.9767	0.9770	0.9774	0.9778	12
78	0.9781	0.9785	0.9789	0.9792	0.9796	0.9799	0.9803	0.9806	0.9810	0.9813	11
79	0.9816	0.9820	0.9823	0.9826	0.9829	0.9833	0.9836	0.9839	0.9842	0.9845	10
80	0.9848	0.9851	0.9854	0.9857	0.9860	0.9863	0.9866	0.9869	0.9871	0.9874	9
81	0.9877	0.9880	0.9882	0.9885	0.9888	0.9890	0.9893	0.9895	0.9898	0.9900	8
82	0.9903	0.9905	0.9907	0.9910	0.9912	0.9914	0.9917	0.9919	0.9921	0.9923	7
83	0.9925	0.9928	0.9930	0.9932	0.9934	0.9936	0.9938	0.9940	0.9942	0.9943	6
84	0.9945	0.9947	0.9949	0.9951	0.9952	0.9954	0.9956	0.9957	0.9959	0.9960	5
85	0.9962	0.9963	0.9965	0.9966	0.9968	0.9969	0.9971	0.9972	0.9973	0.9974	4
86	0.9976	0.9977	0.9978	0.9979	0.9980	0.9981	0.9982	0.9983	0.9984	0.9985	3
87	0.9986	0.9987	0.9988	0.9989	0.9990	0.9990	0.9991	0.9992	0.9993	0.9993	2
88	0.9994	0.9995	0.9995	0.9996	0.9996	0.9997	0.9997	0.9997	0.9998	0.9998	1
89	0.9998	0.9999	0.9999	0.9999	0.9999	1.000	1.000	1.000	1.000	1.000	0
	1.0	0.9	0.8	0.7	0.6	0.5	0.4	0.3	0.2	0.1	Deg

NOTE: For cosines use right-hand column of degrees and lower line of tenths.

Table 1-8. Numbers, Squares, Square Roots, Cubes, and Cube Roots *

N	N^2	$\sqrt{N}$	N^3	$\sqrt[3]{N}$	N	N^2	$\sqrt{N}$	N^3	$\sqrt[3]{N}$
1	1	1.000	1	1.000	40	1,600	6.325	64,000	3.420
2	4	1.414	8	1.260	41	1,681	6.403	68,921	3.448
3	9	1.732	27	1.442	42	1,764	6.481	74,088	3.476
4	16	2.000	64	1.587	43	1,849	6.557	79,507	3.503
5	25	2.236	125	1.710	44	1,936	6.633	85,184	3.530
6	36	2.449	216	1.817	45	2,025	6.708	91,125	3.557
7	49	2.646	343	1.913	46	2,116	6.782	97,336	3.583
8	64	2.828	512	2.000	47	2,209	6.856	103,823	3.609
9	81	3.000	729	2.080	48	2,304	6.928	110,592	3.634
10	100	3.162	1,000	2.154	49	2,401	7.000	117,649	3.659
11	121	3.317	1,331	2.224	50	2,500	7.071	125,000	3.684
12	144	3.464	1,728	2.289	51	2,601	7.141	132,651	3.708
13	169	3.606	2,197	2.351	52	2,704	7.211	140,608	3.733
14	196	3.742	2,744	2.410	53	2,809	7.280	148,877	3.756
15	225	3.873	3,375	2.466	54	2,916	7.348	157,464	3.780
16	256	4.000	4,096	2.520	55	3,025	7.416	166,375	3.803
17	289	4.123	4,913	2.571	56	3,136	7.483	175,616	3.826
18	324	4.243	5,832	2.621	57	3,249	7.550	185,193	3.849
19	361	4.359	6,859	2.668	58	3,364	7.616	195,112	3.871
20	400	4.472	8,000	2.714	59	3,481	7.681	205,379	3.893
21	441	4.583	9,261	2.759	60	3,600	7.746	216,000	3.915
22	484	4.690	10,648	2.802	61	3,721	7.810	226,981	3.936
23	529	4.796	12,167	2.844	62	3,844	7.874	238,328	3.958
24	576	4.899	13,824	2.884	63	3,969	7.937	250,047	3.979
25	625	5.000	15,625	2.924	64	4,096	8.000	262,144	4.000
26	676	5.099	17,576	2.962	65	4,225	8.062	274,625	4.021
27	729	5.196	19,683	3.000	66	4,356	8.124	287,496	4.041
28	784	5.292	21,952	3.037	67	4,489	8.185	300,763	4.062
29	841	5.385	24,389	3.072	68	4,624	8.246	314,432	4.082
30	900	5.477	27,000	3.107	69	4,761	8.307	328,509	4.102
31	961	5.568	29,791	3.141	70	4,900	8.367	343,000	4.121
32	1,024	5.657	32,768	3.175	71	5,041	8.426	357,911	4.141
33	1,089	5.745	35,937	3.208	72	5,184	8.485	373,248	4.160
34	1,156	5.831	39,304	3.240	73	5,329	8.544	389,017	4.179
35	1,225	5.916	42,875	3.271	74	5,476	8.602	405,224	4.198
36	1,296	6.000	46,656	3.302	75	5,625	8.660	421,875	4.217
37	1,369	6.083	50,653	3.332	76	5,776	8.718	438,976	4.236
38	1,444	6.164	54,872	3.362	77	5,929	8.775	456,533	4.254
39	1,521	6.245	59,319	3.391	78	6,084	8.832	474,552	4.273
					79	6,241	8.888	493,039	4.291

* From Perry, J. H. (ed.), "Chemical Engineers' Handbook," 3d ed., McGraw-Hill, 1950.

Table 1-8. Numbers, Squares, Square Roots, Cubes, and Cube Roots (Continued)

N	N^2	$\sqrt{N}$	N^3	$\sqrt[3]{N}$	N	N^2	$\sqrt{N}$	N^3	$\sqrt[3]{N}$
80	6,400	8.944	512,000	4.309	120	14,400	10.954	1,728,000	4.932
81	6,561	9.000	531,441	4.327	121	14,641	11.000	1,771,561	4.946
82	6,724	9.055	551,368	4.344	122	14,884	11.045	1,815,848	4.960
83	6,889	9.110	571,787	4.362	123	15,129	11.091	1,860,867	4.973
84	7,056	9.165	592,704	4.380	124	15,376	11.136	1,906,624	4.987
85	7,225	9.220	614,125	4.397	125	15,625	11.180	1,953,125	5.000
86	7,396	9.274	636,056	4.414	126	15,876	11.225	2,000,376	5.013
87	7,569	9.327	658,503	4.431	127	16,129	11.269	2,048,383	5.027
88	7,744	9.381	681,472	4.448	128	16,384	11.314	2,097,152	5.040
89	7,921	9.434	704,969	4.465	129	16,641	11.358	2,146,689	5.053
90	8,100	9.487	729,000	4.481	130	16,900	11.402	2,197,000	5.066
91	8,281	9.539	753,571	4.498	131	17,161	11.446	2,248,091	5.079
92	8,464	9.592	778,688	4.514	132	17,424	11.489	2,299,968	5.092
93	8,649	9.644	804,357	4.531	133	17,689	11.533	2,352,637	5.104
94	8,836	9.695	830,584	4.547	134	17,956	11.576	2,406,104	5.117
95	9,025	9.747	857,375	4.563	135	18,225	11.619	2,460,375	5.130
96	9,216	9.798	884,736	4.579	136	18,496	11.662	2,515,456	5.143
97	9,409	9.849	912,673	4.595	137	18,769	11.705	2,571,353	5.155
98	9,604	9.899	941,192	4.610	138	19,044	11.747	2,628,072	5.168
99	9,801	9.950	970,299	4.626	139	19,321	11.790	2,685,619	5.180
100	10,000	10.000	1,000,000	4.642	140	19,600	11.832	2,744,000	5.192
101	10,201	10.050	1,030,301	4.657	141	19,881	11.874	2,803,221	5.205
102	10,404	10.100	1,061,208	4.672	142	20,164	11.916	2,863,288	5.217
103	10,609	10.149	1,092,727	4.688	143	20,449	11.958	2,924,207	5.229
104	10,816	10.198	1,124,864	4.703	144	20,736	12.000	2,985,984	5.241
105	11,025	10.247	1,157,625	4.718	145	21,025	12.042	3,048,625	5.254
106	11,236	10.296	1,191,016	4.733	146	21,316	12.083	3,112,136	5.266
107	11,449	10.344	1,225,043	4.747	147	21,609	12.124	3,176,523	5.278
108	11,664	10.392	1,259,712	4.762	148	21,904	12.166	3,241,792	5.290
109	11,881	10.440	1,295,029	4.777	149	22,201	12.207	3,307,949	5.301
110	12,100	10.488	1,331,000	4.791	150	22,500	12.247	3,375,000	5.313
111	12,321	10.536	1,367,631	4.806	151	22,801	12.288	3,442,951	5.325
112	12,544	10.583	1,404,928	4.820	152	23,104	12.329	3,511,808	5.337
113	12,769	10.630	1,442,897	4.835	153	23,409	12.369	3,581,577	5.348
114	12,996	10.677	1,481,544	4.849	154	23,716	12.410	3,652,264	5.360
115	13,225	10.724	1,520,875	4.863	155	24,025	12.450	3,723,875	5.372
116	13,456	10.770	1,560,896	4.877	156	24,336	12.490	3,796,416	5.383
117	13,689	10.817	1,601,613	4.891	157	24,649	12.530	3,869,893	5.395
118	13,924	10.863	1,643,032	4.905	158	24,964	12.570	3,944,312	5.406
119	14,161	10.909	1,685,159	4.919	159	25,281	12.610	4,019,679	5.418

Table 1-8. Numbers, Squares, Square Roots, Cubes, and Cube Roots (Continued)

N	N^2	$\sqrt{N}$	N^3	$\sqrt[3]{N}$	N	N^2	$\sqrt{N}$	N^3	$\sqrt[3]{N}$
160	25,600	12.649	4,096,000	5.429	200	40,000	14.142	8,000,000	5.848
161	25,921	12.689	4,173,281	5.440	201	40,401	14.177	8,120,601	5.858
162	26,244	12.728	4,251,528	5.451	202	40,804	14.213	8,242,408	5.867
163	26,569	12.767	4,330,747	5.463	203	41,209	14.248	8,365,427	5.877
164	26,896	12.806	4,410,944	5.474	204	41,616	14.283	8,489,664	5.887
165	27,225	12.845	4,492,125	5.485	205	42,025	14.318	8,615,125	5.896
166	27,556	12.884	4,574,296	5.496	206	42,436	14.353	8,741,816	5.906
167	27,889	12.923	4,657,463	5.507	207	42,849	14.387	8,869,743	5.915
168	28,224	12.961	4,741,632	5.518	208	43,264	14.422	8,998,912	5.925
169	28,561	13.000	4,826,809	5.529	209	43,681	14.457	9,129,329	5.934
170	28,900	13.038	4,913,000	5.540	210	44,100	14.491	9,261,000	5.944
171	29,241	13.077	5,000,211	5.550	211	44,521	14.526	9,393,931	5.953
172	29,584	13.115	5,088,448	5.561	212	44,944	14.560	9,528,128	5.963
173	29,929	13.153	5,177,717	5.572	213	45,369	14.595	9,663,597	5.972
174	30,276	13.191	5,268,024	5.583	214	45,796	14.629	9,800,344	5.981
175	30,625	13.229	5,359,375	5.593	215	46,225	14.663	9,938,375	5.991
176	30,976	13.266	5,451,776	5.604	216	46,656	14.697	10,077,696	6.000
177	31,329	13.304	5,545,233	5.615	217	47,089	14.731	10,218,313	6.009
178	31,684	13.342	5,639,752	5.625	218	47,524	14.765	10,360,232	6.018
179	32,041	13.379	5,735,339	5.636	219	47,961	14.799	10,503,459	6.028
180	32,400	13.416	5,832,000	5.646	220	48,400	14.832	10,648,000	6.037
181	32,761	13.454	5,929,741	5.657	221	48,841	14.866	10,793,861	6.046
182	33,124	13.491	6,028,568	5.667	222	49,284	14.900	10,941,048	6.055
183	33,489	13.528	6,128,487	5.677	223	49,729	14.933	11,089,567	6.064
184	33,856	13.565	6,229,504	5.688	224	50,176	14.967	11,239,424	6.073
185	34,225	13.601	6,331,625	5.698	225	50,625	15.000	11,390,625	6.082
186	34,596	13.638	6,434,856	5.708	226	51,076	15.033	11,543,176	6.091
187	34,969	13.675	6,539,203	5.718	227	51,529	15.067	11,697,083	6.100
188	35,344	13.711	6,644,672	5.729	228	51,984	15.100	11,852,352	6.109
189	35,721	13.748	6,751,269	5.739	229	52,441	15.133	12,008,989	6.118
190	36,100	13.784	6,859,000	5.749	230	52,900	15.166	12,167,000	6.127
191	36,481	13.820	6,967,871	5.759	231	53,361	15.199	12,326,391	6.136
192	36,864	13.856	7,077,888	5.769	232	53,824	15.232	12,487,168	6.145
193	37,249	13.892	7,189,057	5.779	233	54,289	15.264	12,649,337	6.153
194	37,636	13.928	7,301,384	5.789	234	54,756	15.297	12,812,904	6.162
195	38,025	13.964	7,414,875	5.799	235	55,225	15.330	12,977,875	6.171
196	38,416	14.000	7,529,536	5.809	236	55,696	15.362	13,144,256	6.180
197	38,809	14.036	7,645,373	5.819	237	56,169	15.395	13,312,053	6.188
198	39,204	14.071	7,762,392	5.828	238	56,644	15.427	13,481,272	6.197
199	39,601	14.107	7,880,599	5.838	239	57,121	15.460	13,651,919	6.206

Table 1-8. Numbers, Squares, Square Roots, Cubes, and Cube Roots (Continued)

N	N^2	$\sqrt{N}$	N^3	$\sqrt[3]{N}$	N	N^2	$\sqrt{N}$	N^3	$\sqrt[3]{N}$
240	57,600	15.492	13,824,000	6.214	280	78,400	16.733	21,952,000	6.542
241	58,081	15.524	13,997,521	6.223	281	78,961	16.763	22,188,041	6.550
242	58,564	15.556	14,172,488	6.232	282	79,524	16.793	22,425,768	6.558
243	59,049	15.588	14,348,907	6.240	283	80,089	16.823	22,665,187	6.565
244	59,536	15.620	14,526,784	6.249	284	80,656	16.852	22,906,304	6.573
245	60,025	15.652	14,706,125	6.257	285	81,225	16.882	23,149,125	6.581
246	60,516	15.684	14,886,936	6.266	286	81,796	16.912	23,393,656	6.589
247	61,009	15.716	15,069,223	6.274	287	82,369	16.941	23,639,903	6.596
248	61,504	15.748	15,252,992	6.283	288	82,944	16.971	23,887,872	6.604
249	62,001	15.780	15,438,249	6.291	289	83,521	17.000	24,137,569	6.611
250	62,500	15.811	15,625,000	6.300	290	84,100	17.029	24,389,000	6.619
251	63,001	15.843	15,813,251	6.308	291	84,681	17.059	24,642,171	6.627
252	63,504	15.875	16,003,008	6.316	292	85,264	17.088	24,897,088	6.634
253	64,009	15.906	16,194,277	6.325	293	85,849	17.117	25,153,757	6.642
254	64,516	15.937	16,387,064	6.333	294	86,436	17.146	25,412,184	6.649
255	65,025	15.969	16,581,375	6.341	295	87,025	17.176	25,672,375	6.657
256	65,536	16.000	16,777,216	6.350	296	87,616	17.205	25,934,336	6.664
257	66,049	16.031	16,974,593	6.358	297	88,209	17.234	26,198,073	6.672
258	66,564	16.062	17,173,512	6.366	298	88,804	17.263	26,463,593	6.679
259	67,081	16.093	17,373,979	6.374	299	89,401	17.292	26,730,899	6.687
260	67,600	16.125	17,576,000	6.383	300	90,000	17.321	27,000,000	6.694
261	68,121	16.155	17,779,581	6.391	301	90,601	17.349	27,270,901	6.702
262	68,644	16.186	17,984,728	6.399	302	91,204	17.378	27,543,608	6.709
263	69,169	16.217	18,191,447	6.407	303	91,809	17.407	27,818,127	6.717
264	69,696	16.248	18,399,744	6.415	304	92,416	17.436	28,094,464	6.724
265	70,225	16.279	18,609,625	6.423	305	93,025	17.464	28,372,625	6.731
266	70,756	16.310	18,821,096	6.431	306	93,636	17.493	28,652,616	6.739
267	71,289	16.340	19,034,163	6.439	307	94,249	17.521	28,934,443	6.746
268	71,824	16.371	19,248,832	6.447	308	94,864	17.550	29,218,112	6.753
269	72,361	16.401	19,465,109	6.455	309	95,481	17.578	29,503,629	6.761
270	72,900	16.432	19,683,000	6.463	310	96,100	17.607	29,791,000	6.768
271	73,441	16.462	19,902,511	6.471	311	96,721	17.635	30,080,231	6.775
272	73,984	16.492	20,123,648	6.479	312	97,344	17.664	30,371,328	6.782
273	74,529	16.523	20,346,417	6.487	313	97,969	17.692	30,664,297	6.790
274	75,076	16.553	20,570,824	6.495	314	98,596	17.720	30,959,144	6.797
275	75,625	16.583	20,796,875	6.503	315	99,225	17.748	31,255,875	6.804
276	76,176	16.613	21,024,576	6.511	316	99,856	17.776	31,554,496	6.811
277	76,729	16.643	21,253,933	6.519	317	100,489	17.804	31,855,013	6.818
278	77,284	16.673	21,484,952	6.527	318	101,124	17.833	32,157,432	6.826
279	77,841	16.703	21,717,639	6.534	319	101,761	17.861	32,461,759	6.833

Table 1-8. Numbers, Squares, Square Roots, Cubes, and Cube Roots (Continued)

N	N^2	$\sqrt{N}$	N^3	$\sqrt[3]{N}$	N	N^2	$\sqrt{N}$	N^3	$\sqrt[3]{N}$
320	102,400	17.889	32,768,000	6.840	360	129,600	18.974	46,656,000	7.114
321	103,041	17.916	33,076,161	6.847	361	130,321	19.000	47,045,881	7.120
322	103,684	17.944	33,386,248	6.854	362	131,044	19.026	47,437,928	7.127
323	104,329	17.972	33,698,267	6.861	363	131,769	19.053	47,832,147	7.133
324	104,976	18.000	34,012,224	6.868	364	132,496	19.079	48,228,544	7.140
325	105,625	18.028	34,328,125	6.875	365	133,225	19.105	48,627,125	7.147
326	106,276	18.055	34,645,976	6.882	366	133,956	19.131	49,027,896	7.153
327	106,929	18.083	34,965,783	6.889	367	134,689	19.157	49,430,863	7.160
328	107,584	18.111	35,287,552	6.896	368	135,424	19.183	49,836,032	7.166
329	108,241	18.138	35,611,289	6.903	369	136,161	19.209	50,243,409	7.173
330	108,900	18.166	35,937,000	6.910	370	136,900	19.235	50,653,000	7.179
331	109,561	18.193	36,264,691	6.917	371	137,641	19.261	51,064,811	7.186
332	110,224	18.221	36,594,368	6.924	372	138,384	19.287	51,478,848	7.192
333	110,889	18.248	36,926,037	6.931	373	139,129	19.313	51,895,117	7.198
334	111,556	18.276	37,259,704	6.938	374	139,876	19.339	52,313,624	7.205
335	112,225	18.303	37,595,375	6.945	375	140,625	19.365	52,734,375	7.211
336	112,896	18.330	37,933,056	6.952	376	141,376	19.391	53,157,376	7.218
337	113,569	18.358	38,272,753	6.959	377	142,129	19.416	53,582,633	7.224
338	114,244	18.385	38,614,472	6.966	378	142,884	19.442	54,010,152	7.230
339	114,921	18.412	38,958,219	6.973	379	143,641	19.468	54,439,939	7.237
340	115,600	18.439	39,304,000	6.980	380	144,400	19.494	54,872,000	7.243
341	116,281	18.466	39,651,821	6.986	381	145,161	19.519	55,306,341	7.250
342	116,964	18.493	40,001,688	6.993	382	145,924	19.545	55,742,968	7.256
343	117,649	18.520	40,353,607	7.000	383	146,689	19.570	56,181,887	7.262
344	118,336	18.547	40,707,584	7.007	384	147,456	19.596	56,623,104	7.268
345	119,025	18.574	41,063,625	7.014	385	148,225	19.621	57,066,625	7.275
346	119,716	18.601	41,421,736	7.020	386	148,996	19.647	57,512,456	7.281
347	120,409	18.628	41,781,923	7.027	387	149,769	19.672	57,960,603	7.287
348	121,104	18.655	42,144,192	7.034	388	150,544	19.698	58,411,072	7.294
349	121,801	18.682	42,508,549	7.041	389	151,321	19.723	58,863,869	7.300
350	122,500	18.708	42,875,000	7.047	390	152,100	19.748	59,319,000	7.306
351	123,201	18.735	43,243,551	7.054	391	152,881	19.774	59,776,471	7.312
352	123,904	18.762	43,614,208	7.061	392	153,664	19.799	60,236,288	7.319
353	124,609	18.788	43,986,977	7.067	393	154,449	19.824	60,698,457	7.325
354	125,316	18.815	44,361,864	7.074	394	155,236	19.849	61,162,984	7.331
355	126,025	18.841	44,738,875	7.081	395	156,025	19.875	61,629,875	7.337
356	126,736	18.868	45,118,016	7.087	396	156,816	19.900	62,099,136	7.343
357	127,449	18.894	45,499,293	7.094	397	157,609	19.925	62,570,773	7.350
358	128,164	18.921	45,882,712	7.101	398	158,404	19.950	63,044,792	7.356
359	128,881	18.947	46,268,279	7.107	399	159,201	19.975	63,521,199	7.362

Table 1-8. Numbers, Squares, Square Roots, Cubes, and Cube Roots (Continued)

<i>N</i>	<i>N</i> ²	$\sqrt{N}$	<i>N</i> ³	$\sqrt[3]{N}$	<i>N</i>	<i>N</i> ²	$\sqrt{N}$	<i>N</i> ³	$\sqrt[3]{N}$
400	160,000	20.000	64,000,000	7.368	440	193,600	20.976	85,184,000	7.606
401	160,801	20.025	64,481,201	7.374	441	194,481	21.000	85,766,121	7.612
402	161,604	20.050	64,964,808	7.380	442	195,364	21.024	86,350,888	7.617
403	162,409	20.075	65,450,827	7.386	443	196,249	21.048	86,938,307	7.623
404	163,216	20.100	65,939,264	7.393	444	197,136	21.071	87,528,384	7.629
405	164,025	20.125	66,430,125	7.399	445	198,025	21.095	88,121,125	7.635
406	164,836	20.149	66,923,416	7.405	446	198,916	21.119	88,716,536	7.640
407	165,649	20.174	67,419,143	7.411	447	199,809	21.142	89,314,623	7.646
408	166,464	20.199	67,917,312	7.417	448	200,704	21.166	89,915,392	7.652
409	167,281	20.224	68,417,929	7.423	449	201,601	21.190	90,518,849	7.657
410	168,100	20.248	68,921,000	7.429	450	202,500	21.213	91,125,000	7.663
411	168,921	20.273	69,426,531	7.435	451	203,401	21.237	91,733,851	7.669
412	169,744	20.298	69,934,528	7.441	452	204,304	21.260	92,345,408	7.674
413	170,569	20.322	70,444,997	7.447	453	205,209	21.284	92,959,677	7.680
414	171,396	20.347	70,957,944	7.453	454	206,116	21.307	93,576,664	7.686
415	172,225	20.372	71,473,375	7.459	455	207,025	21.331	94,196,375	7.691
416	173,056	20.396	71,991,296	7.465	456	207,936	21.354	94,818,816	7.697
417	173,889	20.421	72,511,713	7.471	457	208,849	21.378	95,443,993	7.703
418	174,724	20.445	73,034,632	7.477	458	209,764	21.401	96,071,912	7.708
419	175,561	20.469	73,560,059	7.483	459	210,681	21.424	96,702,579	7.714
420	176,400	20.494	74,088,000	7.489	460	211,600	21.448	97,336,000	7.719
421	177,241	20.518	74,618,461	7.495	461	212,521	21.471	97,972,181	7.725
422	178,084	20.543	75,151,448	7.501	462	213,444	21.494	98,611,128	7.731
423	178,929	20.567	75,686,967	7.507	463	214,369	21.517	99,252,847	7.736
424	179,776	20.591	76,225,024	7.513	464	215,296	21.541	99,897,344	7.742
425	180,625	20.616	76,765,625	7.518	465	216,225	21.564	100,544,625	7.747
426	181,476	20.640	77,308,776	7.524	466	217,156	21.587	101,194,696	7.753
427	182,329	20.664	77,854,483	7.530	467	218,089	21.610	101,847,563	7.758
428	183,184	20.688	78,402,752	7.536	468	219,024	21.633	102,503,232	7.764
429	184,041	20.712	78,953,589	7.542	469	219,961	21.656	103,161,709	7.769
430	184,900	20.736	79,507,000	7.548	470	220,900	21.679	103,823,000	7.775
431	185,761	20.761	80,062,991	7.554	471	221,841	21.703	104,487,111	7.780
432	186,624	20.785	80,621,568	7.560	472	222,784	21.726	105,154,048	7.786
433	187,489	20.809	81,182,737	7.565	473	223,729	21.749	105,823,817	7.791
434	188,356	20.833	81,746,504	7.571	474	224,676	21.772	106,496,424	7.797
435	189,225	20.857	82,312,875	7.577	475	225,625	21.794	107,171,875	7.802
436	190,096	20.881	82,881,856	7.583	476	226,576	21.817	107,850,176	7.808
437	190,969	20.905	83,453,453	7.589	477	227,529	21.840	108,531,333	7.813
438	191,844	20.928	84,027,672	7.594	478	228,484	21.863	109,215,352	7.819
439	192,721	20.952	84,604,519	7.600	479	229,441	21.886	109,902,239	7.824

Table 1-8. Numbers, Squares, Square Roots, Cubes, and Cube Roots (Continued)

N	N^2	$\sqrt{N}$	N^3	$\sqrt[3]{N}$	N	N^2	$\sqrt{N}$	N^3	$\sqrt[3]{N}$
480	230,400	21.909	110,592,000	7.830	520	270,400	22.804	140,608,000	8.041
481	231,361	21.932	111,284,641	7.835	521	271,441	22.825	141,420,761	8.047
482	232,324	21.954	111,980,168	7.841	522	272,484	22.847	142,236,648	8.052
483	233,289	21.977	112,678,587	7.846	523	273,529	22.869	143,055,667	8.057
484	234,256	22.000	113,379,904	7.851	524	274,576	22.891	143,877,824	8.062
485	235,225	22.023	114,084,125	7.857	525	275,625	22.913	144,703,125	8.067
486	236,196	22.045	114,791,256	7.862	526	276,676	22.935	145,531,576	8.072
487	237,169	22.068	115,501,303	7.868	527	277,729	22.956	146,363,183	8.077
488	238,144	22.091	116,214,272	7.873	528	278,784	22.978	147,197,952	8.082
489	239,121	22.113	116,930,169	7.878	529	279,841	23.000	148,035,889	8.088
490	240,100	22.136	117,649,000	7.884	530	280,900	23.022	148,877,000	8.093
491	241,081	22.159	118,370,771	7.889	531	281,961	23.043	149,721,291	8.098
492	242,064	22.181	119,095,488	7.894	532	283,024	23.065	150,568,768	8.103
493	243,049	22.204	119,823,157	7.900	533	284,089	23.087	151,419,437	8.108
494	244,036	22.226	120,553,784	7.905	534	285,156	23.108	152,273,304	8.113
495	245,025	22.249	121,287,375	7.910	535	286,225	23.130	153,130,375	8.118
496	246,016	22.271	122,023,936	7.916	536	287,296	23.152	153,990,656	8.123
497	247,009	22.293	122,763,473	7.921	537	288,369	23.173	154,854,153	8.128
498	248,004	22.316	123,505,992	7.926	538	289,444	23.195	155,720,872	8.133
499	249,001	22.338	124,251,499	7.932	539	290,521	23.216	156,590,819	8.138
500	250,000	22.361	125,000,000	7.937	540	291,600	23.238	157,464,000	8.143
501	251,001	22.383	125,751,501	7.942	541	292,681	23.259	158,340,421	8.148
502	252,004	22.405	126,506,008	7.948	542	293,764	23.281	159,220,088	8.153
503	253,009	22.428	127,263,527	7.953	543	294,849	23.302	160,103,007	8.158
504	254,016	22.450	128,024,064	7.958	544	295,936	23.324	160,989,184	8.163
505	255,025	22.472	128,787,625	7.963	545	297,025	23.345	161,878,625	8.168
506	256,036	22.494	129,554,216	7.969	546	298,116	23.367	162,771,336	8.173
507	257,049	22.517	130,323,843	7.974	547	299,209	23.388	163,667,323	8.178
508	258,064	22.539	131,096,512	7.979	548	300,304	23.409	164,566,592	8.183
509	259,081	22.561	131,872,229	7.984	549	301,401	23.431	165,469,149	8.188
510	260,100	22.583	132,651,000	7.990	550	302,500	23.452	166,375,000	8.193
511	261,121	22.605	133,432,831	7.995	551	303,601	23.473	167,284,151	8.198
512	262,144	22.627	134,217,728	8.000	552	304,704	23.495	168,196,608	8.203
513	263,169	22.650	135,005,697	8.005	553	305,809	23.516	169,112,377	8.208
514	264,196	22.672	135,796,744	8.010	554	306,916	23.537	170,031,464	8.213
515	265,225	22.694	136,590,875	8.016	555	308,025	23.558	170,953,875	8.218
516	266,256	22.716	137,388,096	8.021	556	309,136	23.580	171,879,616	8.223
517	267,289	22.738	138,188,413	8.026	557	310,249	23.601	172,808,693	8.228
518	268,324	22.760	138,991,832	8.031	558	311,364	23.622	173,741,112	8.233
519	269,361	22.782	139,798,359	8.036	559	312,481	23.643	174,676,879	8.238

Table 1-8. Numbers, Squares, Square Roots, Cubes, and Cube Roots (Continued)

N	N^2	$\sqrt{N}$	N^3	$\sqrt[3]{N}$	N	N^2	$\sqrt{N}$	N^3	$\sqrt[3]{N}$
560	313,600	23.664	175,616,000	8.243	600	360,000	24.495	216,000,000	8.434
561	314,721	23.685	176,558,481	8.247	601	361,201	24.515	217,081,801	8.439
562	315,844	23.707	177,504,328	8.252	602	362,404	24.536	218,167,208	8.444
563	316,969	23.728	178,453,547	8.257	603	363,609	24.556	219,256,227	8.448
564	318,096	23.749	179,406,144	8.262	604	364,816	24.576	220,348,864	8.453
565	319,225	23.770	180,362,125	8.267	605	366,025	24.597	221,445,125	8.458
566	320,356	23.791	181,321,496	8.272	606	367,236	24.617	222,545,016	8.462
567	321,489	23.812	182,284,263	8.277	607	368,449	24.637	223,648,543	8.467
568	322,624	23.833	183,250,432	8.282	608	369,664	24.658	224,755,712	8.472
569	323,761	23.854	184,220,009	8.286	609	370,881	24.678	225,866,529	8.476
570	324,900	23.875	185,193,000	8.291	610	372,100	24.698	226,981,000	8.481
571	326,041	23.896	186,169,411	8.296	611	373,321	24.718	228,099,131	8.486
572	327,184	23.917	187,149,248	8.301	612	374,544	24.739	229,220,928	8.490
573	328,329	23.937	188,132,517	8.306	613	375,769	24.759	230,346,397	8.495
574	329,476	23.958	189,119,224	8.311	614	376,996	24.779	231,475,544	8.499
575	330,625	23.979	190,109,375	8.316	615	378,225	24.799	232,608,375	8.504
576	331,776	24.000	191,102,976	8.320	616	379,456	24.819	233,744,896	8.509
577	332,929	24.021	192,100,033	8.325	617	380,689	24.839	234,885,113	8.513
578	334,084	24.042	193,100,552	8.330	618	381,924	24.860	236,029,032	8.518
579	335,241	24.062	194,104,539	8.335	619	383,161	24.880	237,176,659	8.522
580	336,400	24.083	195,112,000	8.340	620	384,400	24.900	238,328,000	8.527
581	337,561	24.104	196,122,941	8.344	621	385,641	24.920	239,483,061	8.532
582	338,724	24.125	197,137,368	8.349	622	386,884	24.940	240,641,848	8.536
583	339,889	24.145	198,155,287	8.354	623	388,129	24.960	241,804,367	8.541
584	341,056	24.166	199,176,704	8.359	624	389,376	24.980	242,970,624	8.545
585	342,225	24.187	200,201,625	8.363	625	390,625	25.000	244,140,625	8.550
586	343,396	24.207	201,230,056	8.368	626	391,876	25.020	245,314,376	8.554
587	344,569	24.228	202,262,003	8.373	627	393,129	25.040	246,491,883	8.559
588	345,744	24.249	203,297,472	8.378	628	394,384	25.060	247,673,152	8.564
589	346,921	24.269	204,336,469	8.382	629	395,641	25.080	248,858,189	8.568
590	348,100	24.290	205,379,000	8.387	630	396,900	25.100	250,047,000	8.573
591	349,281	24.310	206,425,071	8.392	631	398,161	25.120	251,239,591	8.577
592	350,464	24.331	207,474,688	8.397	632	399,424	25.140	252,435,968	8.582
593	351,649	24.352	208,527,857	8.401	633	400,689	25.159	253,636,137	8.586
594	352,836	24.372	209,584,584	8.406	634	401,956	25.179	254,840,104	8.591
595	354,025	24.393	210,644,875	8.411	635	403,225	25.199	256,047,875	8.595
596	355,216	24.413	211,708,736	8.416	636	404,496	25.219	257,259,456	8.600
597	356,409	24.434	212,776,173	8.420	637	405,769	25.239	258,474,853	8.604
598	357,604	24.454	213,847,192	8.425	638	407,044	25.259	259,694,072	8.609
599	358,801	24.474	214,921,799	8.430	639	408,321	25.278	260,917,119	8.613

Table 1-8. Numbers, Squares, Square Roots, Cubes, and Cube Roots (Continued)

N	N^2	$\sqrt{N}$	N^3	$\sqrt[3]{N}$	N	N^2	$\sqrt{N}$	N^3	$\sqrt[3]{N}$
640	409,600	25.298	262,144,000	8.618	680	462,400	26.077	314,432,000	8.794
641	410,881	25.318	263,374,721	8.622	681	463,761	26.096	315,821,241	8.798
642	412,164	25.338	264,609,288	8.627	682	465,124	26.115	317,214,568	8.802
643	413,449	25.357	265,847,707	8.631	683	466,489	26.134	318,611,987	8.807
644	414,736	25.377	267,089,984	8.636	684	467,856	26.153	320,013,504	8.811
645	416,025	25.397	268,336,125	8.640	685	469,225	26.173	321,419,125	8.815
646	417,316	25.417	269,586,136	8.645	686	470,596	26.192	322,828,856	8.819
647	418,609	25.436	270,840,023	8.649	687	471,969	26.211	324,242,703	8.824
648	419,904	25.456	272,097,792	8.653	688	473,344	26.230	325,660,672	8.828
649	421,201	25.475	273,359,449	8.658	689	474,721	26.249	327,082,769	8.832
650	422,500	25.495	274,625,000	8.662	690	476,100	26.268	328,509,000	8.837
651	423,801	25.515	275,894,451	8.667	691	477,481	26.287	329,939,371	8.841
652	425,104	25.534	277,167,808	8.671	692	478,864	26.306	331,373,888	8.845
653	426,409	25.554	278,445,077	8.676	693	480,249	26.325	332,812,557	8.849
654	427,716	25.573	279,726,264	8.680	694	481,636	26.344	334,255,384	8.854
655	429,025	25.593	281,011,375	8.685	695	483,025	26.363	335,702,375	8.858
656	430,336	25.612	282,300,416	8.689	696	484,416	26.382	337,153,536	8.862
657	431,649	25.632	283,593,393	8.693	697	485,809	26.401	338,608,873	8.866
658	432,964	25.652	284,890,312	8.698	698	487,204	26.420	340,068,392	8.871
659	434,281	25.671	286,191,179	8.702	699	488,601	26.439	341,532,099	8.875
660	435,600	25.690	287,496,000	8.707	700	490,000	26.458	343,000,000	8.879
661	436,921	25.710	288,804,781	8.711	701	491,401	26.476	344,472,101	8.883
662	438,244	25.729	290,117,528	8.715	702	492,804	26.495	345,948,408	8.887
663	439,569	25.749	291,434,247	8.720	703	494,209	26.514	347,428,927	8.892
664	440,896	25.768	292,754,944	8.724	704	495,616	26.533	348,913,664	8.896
665	442,225	25.788	294,079,625	8.729	705	497,025	26.552	350,402,625	8.900
666	443,556	25.807	295,408,296	8.733	706	498,436	26.571	351,895,816	8.904
667	444,889	25.826	296,740,963	8.737	707	499,849	26.589	353,393,243	8.909
668	446,224	25.846	298,077,632	8.742	708	501,264	26.608	354,894,912	8.913
669	447,561	25.865	299,418,309	8.746	709	502,681	26.627	356,400,829	8.917
670	448,900	25.884	300,763,000	8.750	710	504,100	26.646	357,911,000	8.921
671	450,241	25.904	302,111,711	8.755	711	505,521	26.665	359,425,431	8.925
672	451,584	25.923	303,464,448	8.759	712	506,944	26.683	360,944,128	8.929
673	452,929	25.942	304,821,217	8.763	713	508,369	26.702	362,467,097	8.934
674	454,276	25.962	306,182,024	8.768	714	509,796	26.721	363,994,344	8.938
675	455,625	25.981	307,546,875	8.772	715	511,225	26.739	365,525,875	8.942
676	456,976	26.000	308,915,776	8.776	716	512,656	26.758	367,061,696	8.946
677	458,329	26.019	310,288,733	8.781	717	514,089	26.777	368,601,813	8.950
678	459,684	26.038	311,665,752	8.785	718	515,524	26.796	370,146,232	8.955
679	461,041	26.058	313,046,839	8.789	719	516,961	26.814	371,694,959	8.959

Table 1-8. Numbers, Squares, Square Roots, Cubes, and Cube Roots (Continued)

N	N^2	$\sqrt{N}$	N^3	$\sqrt[3]{N}$	N	N^2	$\sqrt{N}$	N^3	$\sqrt[3]{N}$
720	518,400	26.833	373,248,000	8.963	760	577,600	27.568	438,976,000	9.126
721	519,841	26.851	374,805,361	8.967	761	579,121	27.586	440,711,081	9.130
722	521,284	26.870	376,367,048	8.971	762	580,644	27.604	442,450,728	9.134
723	522,729	26.889	377,933,067	8.975	763	582,169	27.622	444,194,947	9.138
724	524,176	26.907	379,503,424	8.979	764	583,696	27.641	445,943,744	9.142
725	525,625	26.926	381,078,125	8.984	765	585,225	27.659	447,697,125	9.146
726	527,076	26.944	382,657,176	8.988	766	586,756	27.677	449,455,096	9.150
727	528,529	26.963	384,240,583	8.992	767	588,289	27.695	451,217,663	9.154
728	529,984	26.981	385,828,352	8.996	768	589,824	27.713	452,984,832	9.158
729	531,441	27.000	387,420,489	9.000	769	591,361	27.731	454,756,609	9.162
730	532,900	27.019	389,017,000	9.004	770	592,900	27.749	456,533,000	9.166
731	534,361	27.037	390,617,891	9.008	771	594,441	27.767	458,314,011	9.170
732	535,824	27.055	392,223,168	9.012	772	595,984	27.785	460,099,648	9.174
733	537,289	27.074	393,832,837	9.016	773	597,529	27.803	461,889,917	9.178
734	538,756	27.092	395,446,904	9.021	774	599,076	27.821	463,684,824	9.182
735	540,225	27.111	397,065,375	9.025	775	600,625	27.839	465,484,375	9.185
736	541,696	27.129	398,688,256	9.029	776	602,176	27.857	467,288,576	9.189
737	543,169	27.148	400,315,553	9.033	777	603,729	27.875	469,097,433	9.193
738	544,644	27.166	401,947,272	9.037	778	605,284	27.893	470,910,952	9.197
739	546,121	27.185	403,583,419	9.041	779	606,841	27.911	472,729,139	9.201
740	547,600	27.203	405,224,000	9.045	780	608,400	27.928	474,552,000	9.205
741	549,081	27.221	406,869,021	9.049	781	609,961	27.946	476,379,541	9.209
742	550,564	27.240	408,518,488	9.053	782	611,524	27.964	478,211,768	9.213
743	552,049	27.258	410,172,407	9.057	783	613,089	27.982	480,048,687	9.217
744	553,536	27.276	411,830,784	9.061	784	614,656	28.000	481,890,304	9.221
745	555,025	27.295	413,493,625	9.065	785	616,225	28.018	483,736,625	9.225
746	556,516	27.313	415,160,936	9.069	786	617,796	28.036	485,587,656	9.229
747	558,009	27.331	416,832,723	9.073	787	619,369	28.054	487,443,403	9.233
748	559,504	27.350	418,508,992	9.078	788	620,944	28.071	489,303,872	9.237
749	561,001	27.368	420,189,749	9.082	789	622,521	28.089	491,169,069	9.240
750	562,500	27.386	421,875,000	9.086	790	624,100	28.107	493,039,000	9.244
751	564,001	27.404	423,564,751	9.090	791	625,681	28.125	494,913,671	9.248
752	565,504	27.423	425,259,008	9.094	792	627,264	28.142	496,793,088	9.252
753	567,009	27.441	426,957,777	9.098	793	628,849	28.160	498,677,257	9.256
754	568,516	27.459	428,661,064	9.102	794	630,436	28.178	500,566,184	9.260
755	570,025	27.477	430,368,875	9.106	795	632,025	28.196	502,459,875	9.264
756	571,536	27.495	432,081,216	9.110	796	633,616	28.213	504,358,336	9.268
757	573,049	27.514	433,798,093	9.114	797	635,209	28.231	506,261,573	9.272
758	574,564	27.532	435,519,512	9.118	798	636,804	28.249	508,169,592	9.275
759	576,081	27.550	437,245,479	9.122	799	638,401	28.267	510,082,399	9.279

Table 1-8. Numbers, Squares, Square Roots, Cubes, and Cube Roots (Continued)

N	N^2	$\sqrt{N}$	N^3	$\sqrt[3]{N}$	N	N^2	$\sqrt{N}$	N^3	$\sqrt[3]{N}$
800	640,000	28.284	512,000,000	9.283	840	705,600	28.983	592,704,000	9.435
801	641,601	28.302	513,922,401	9.287	841	707,281	29.000	594,823,321	9.439
802	643,204	28.320	515,849,608	9.291	842	708,964	29.017	596,947,688	9.443
803	644,809	28.337	517,781,627	9.295	843	710,649	29.034	599,077,107	9.447
804	646,416	28.355	519,718,464	9.299	844	712,236	29.052	601,211,584	9.450
805	648,025	28.373	521,660,125	9.302	845	714,025	29.069	603,351,125	9.454
806	649,636	28.390	523,606,616	9.306	846	715,716	29.086	605,495,736	9.458
807	651,249	28.408	525,557,943	9.310	847	717,409	29.103	607,645,423	9.462
808	652,864	28.425	527,514,112	9.314	848	719,104	29.120	609,800,192	9.465
809	654,481	28.443	529,475,129	9.318	849	720,801	29.138	611,960,049	9.469
810	656,100	28.460	531,441,000	9.322	850	722,500	29.155	614,125,000	9.473
811	657,721	28.478	533,411,731	9.326	851	724,201	29.172	616,295,051	9.476
812	659,344	28.496	535,387,328	9.329	852	725,904	29.189	618,470,208	9.480
813	660,969	28.513	537,367,797	9.333	853	727,609	29.206	620,650,477	9.484
814	662,596	28.531	539,353,144	9.337	854	729,316	29.223	622,835,864	9.488
815	664,225	28.548	541,343,375	9.341	855	731,025	29.240	625,026,375	9.491
816	665,856	28.566	543,338,496	9.345	856	732,736	29.257	627,222,016	9.495
817	667,489	28.583	545,338,513	9.348	857	734,449	29.275	629,422,793	9.499
818	669,124	28.601	547,343,432	9.352	858	736,164	29.292	631,628,712	9.502
819	670,761	28.618	549,353,259	9.356	859	737,881	29.309	633,839,779	9.506
820	672,400	28.636	551,368,000	9.360	860	739,600	29.326	636,056,000	9.510
821	674,041	28.653	553,387,661	9.364	861	741,321	29.343	638,277,381	9.513
822	675,684	28.671	555,412,248	9.368	862	743,044	29.360	640,503,928	9.517
823	677,329	28.688	557,441,767	9.371	863	744,769	29.377	642,735,647	9.521
824	678,976	28.705	559,476,224	9.375	864	746,496	29.394	644,972,544	9.524
825	680,625	28.723	561,515,625	9.379	865	748,225	29.411	647,214,625	9.528
826	682,276	28.740	563,559,976	9.383	866	749,956	29.428	649,461,896	9.532
827	683,929	28.758	565,609,283	9.386	867	751,689	29.445	651,714,363	9.535
828	685,584	28.775	567,663,552	9.390	868	753,424	29.462	653,972,032	9.539
829	687,241	28.792	569,722,789	9.394	869	755,161	29.479	656,234,909	9.543
830	688,900	28.810	571,787,000	9.398	870	756,900	29.496	658,503,000	9.546
831	690,561	28.827	573,856,191	9.402	871	758,641	29.513	660,776,311	9.550
832	692,224	28.844	575,930,368	9.405	872	760,384	29.530	663,054,848	9.554
833	693,889	28.862	578,009,537	9.409	873	762,129	29.547	665,338,617	9.557
834	695,556	28.879	580,093,704	9.413	874	763,876	29.563	667,627,624	9.561
835	697,225	28.896	582,182,875	9.417	875	765,625	29.580	669,921,875	9.565
836	698,896	28.914	584,277,056	9.420	876	767,376	29.597	672,221,376	9.568
837	700,569	28.931	586,376,253	9.424	877	769,129	29.614	674,526,133	9.572
838	702,244	28.948	588,480,472	9.428	878	770,884	29.631	676,836,152	9.576
839	703,921	28.965	590,589,719	9.432	879	772,641	29.648	679,151,439	9.579

Table 1-8. Numbers, Squares, Square Roots, Cubes, and Cube Roots (Continued)

N	N^2	$\sqrt{N}$	N^3	$\sqrt[3]{N}$	N	N^2	$\sqrt{N}$	N^3	$\sqrt[3]{N}$
880	774,400	29.665	681,472,000	9.583	920	846,400	30.332	778,688,000	9.726
881	776,161	29.682	683,797,841	9.586	921	848,241	30.348	781,229,961	9.729
882	777,924	29.698	686,128,968	9.590	922	850,084	30.364	783,777,448	9.733
883	779,689	29.715	688,465,387	9.594	923	851,929	30.381	786,330,467	9.736
884	781,456	29.732	690,807,104	9.597	924	853,776	30.397	788,889,024	9.740
885	783,225	29.749	693,154,125	9.601	925	855,625	30.414	791,453,125	9.743
886	784,996	29.766	695,506,456	9.605	926	857,476	30.430	794,022,776	9.747
887	786,769	29.783	697,864,103	9.608	927	859,329	30.447	796,597,983	9.750
888	788,544	29.799	700,227,072	9.612	928	861,184	30.463	799,178,752	9.754
889	790,321	29.816	702,595,369	9.615	929	863,041	30.480	801,765,089	9.758
890	792,100	29.833	704,969,000	9.619	930	864,900	30.496	804,357,000	9.761
891	793,881	29.850	707,347,971	9.623	931	866,761	30.512	806,954,491	9.764
892	795,664	29.866	709,732,288	9.626	932	868,624	30.529	809,557,568	9.768
893	797,449	29.883	712,121,957	9.630	933	870,489	30.545	812,166,237	9.771
894	799,236	29.900	714,516,984	9.633	934	872,356	30.561	814,780,504	9.775
895	801,025	29.917	716,917,375	9.637	935	874,225	30.578	817,400,375	9.778
896	802,816	29.933	719,323,136	9.641	936	876,096	30.594	820,025,856	9.782
897	804,609	29.950	721,734,273	9.644	937	877,969	30.610	822,656,953	9.785
898	806,404	29.967	724,150,792	9.648	938	879,844	30.627	825,293,672	9.789
899	808,201	29.983	726,572,699	9.651	939	881,721	30.643	827,936,019	9.792
900	810,000	30.000	729,000,000	9.655	940	883,600	30.659	830,584,000	9.796
901	811,801	30.017	731,432,701	9.658	941	885,481	30.676	833,237,621	9.799
902	813,604	30.033	733,870,808	9.662	942	887,364	30.692	835,896,888	9.803
903	815,409	30.050	736,314,327	9.666	943	889,249	30.708	838,561,807	9.806
904	817,216	30.067	738,763,264	9.669	944	891,136	30.725	841,232,384	9.810
905	819,025	30.083	741,217,625	9.673	945	893,025	30.741	843,908,625	9.813
906	820,836	30.100	743,677,416	9.676	946	894,916	30.757	846,590,536	9.817
907	822,649	30.116	746,142,643	9.680	947	896,809	30.773	849,278,123	9.820
908	824,464	30.133	748,613,312	9.683	948	898,704	30.790	851,971,392	9.824
909	826,281	30.150	751,089,429	9.687	949	900,601	30.806	854,670,349	9.827
910	828,100	30.166	753,571,000	9.691	950	902,500	30.822	857,375,000	9.830
911	829,921	30.183	756,058,031	9.694	951	904,401	30.838	860,085,351	9.834
912	831,744	30.199	758,550,528	9.698	952	906,304	30.854	862,801,408	9.837
913	833,569	30.216	761,048,497	9.701	953	908,209	30.871	865,523,177	9.841
914	835,396	30.232	763,551,944	9.705	954	910,116	30.887	868,250,664	9.844
915	837,225	30.249	766,060,875	9.708	955	912,025	30.903	870,983,875	9.848
916	839,056	30.265	768,575,296	9.712	956	913,936	30.919	873,722,816	9.851
917	840,889	30.282	771,095,213	9.715	957	915,849	30.935	876,467,493	9.855
918	842,724	30.299	773,620,632	9.719	958	917,764	30.952	879,217,912	9.858
919	844,561	30.315	776,151,559	9.722	959	919,681	30.968	881,974,079	9.861

Table 1-8. Numbers, Squares, Square Roots, Cubes, and Cube Roots (Continued)

N	N^2	$\sqrt{N}$	N^3	$\sqrt[3]{N}$	N	N^2	$\sqrt{N}$	N^3	$\sqrt[3]{N}$
960	921,600	30.984	884,736,000	9.865	980	960,400	31.305	941,192,000	9.933
961	923,521	31.000	887,503,681	9.868	981	962,361	31.321	944,076,141	9.936
962	925,444	31.016	890,277,128	9.872	982	964,324	31.337	946,966,168	9.940
963	927,369	31.032	893,056,347	9.875	983	966,289	31.353	949,862,087	9.943
964	929,296	31.048	895,841,344	9.879	984	968,256	31.369	952,763,904	9.946
965	931,225	31.064	898,632,125	9.882	985	970,225	31.385	955,671,625	9.950
966	933,156	31.081	901,428,696	9.885	986	972,196	31.401	958,585,256	9.953
967	935,089	31.097	904,231,063	9.889	987	974,169	31.417	961,504,803	9.956
968	937,024	31.113	907,039,232	9.892	988	976,144	31.432	964,430,272	9.960
969	938,961	31.129	909,853,209	9.896	989	978,121	31.448	967,361,669	9.963
970	940,900	31.145	912,673,000	9.899	990	980,100	31.464	970,299,000	9.967
971	942,841	31.161	915,498,611	9.902	991	982,081	31.480	973,242,271	9.970
972	944,784	31.177	918,330,048	9.906	992	984,064	31.496	976,191,488	9.973
973	946,729	31.193	921,167,317	9.909	993	986,049	31.512	979,146,657	9.977
974	948,676	31.209	924,010,424	9.913	994	988,036	31.528	982,107,784	9.980
975	950,625	31.225	926,859,375	9.916	995	990,025	31.544	985,074,875	9.983
976	952,576	31.241	929,714,176	9.919	996	992,016	31.559	988,047,936	9.987
977	954,529	31.257	932,574,833	9.923	997	994,009	31.575	991,026,973	9.990
978	956,484	31.273	935,441,352	9.926	998	996,004	31.591	994,011,992	9.993
979	958,441	31.289	938,313,739	9.930	999	998,001	31.607	997,002,999	9.997

Table 1-9. Values and Logarithms of Exponential Functions

x	e^x		e^{-x} value	x	e^x		e^{-x} value
	Value	$\log_{10}$			Value	$\log_{10}$	
0.00	1.0000	0.00000	1.00000	0.35	1.4191	0.15200	0.70469
0.01	1.0101	0.00434	0.99005	0.36	1.4333	0.15635	0.69768
0.02	1.0202	0.00869	0.98020	0.37	1.4477	0.16069	0.69073
0.03	1.0305	0.01303	0.97045	0.38	1.4623	0.16503	0.68386
0.04	1.0408	0.01737	0.96079	0.39	1.4770	0.16937	0.67706
0.05	1.0513	0.02171	0.95123	0.40	1.4918	0.17372	0.67032
0.06	1.0618	0.02606	0.94176	0.41	1.5068	0.17806	0.66365
0.07	1.0725	0.03040	0.93239	0.42	1.5220	0.18240	0.65705
0.08	1.0833	0.03474	0.92312	0.43	1.5373	0.18675	0.65051
0.09	1.0942	0.03909	0.91393	0.44	1.5527	0.19109	0.64404
0.10	1.1052	0.04343	0.90484	0.45	1.5683	0.19543	0.63763
0.11	1.1163	0.04777	0.89583	0.46	1.5841	0.19978	0.63128
0.12	1.1275	0.05212	0.88692	0.47	1.6000	0.20412	0.62500
0.13	1.1388	0.05646	0.87809	0.48	1.6161	0.20846	0.61878
0.14	1.1503	0.06080	0.86936	0.49	1.6323	0.21280	0.61263
0.15	1.1618	0.06514	0.86071	0.50	1.6487	0.21715	0.60653
0.16	1.1735	0.06949	0.85214	0.51	1.6653	0.22149	0.60050
0.17	1.1853	0.07383	0.84366	0.52	1.6820	0.22583	0.59452
0.18	1.1972	0.07817	0.83527	0.53	1.6989	0.23018	0.58860
0.19	1.2092	0.08252	0.82696	0.54	1.7160	0.23452	0.58275
0.20	1.2214	0.08686	0.81873	0.55	1.7333	0.23886	0.57695
0.21	1.2337	0.09120	0.81058	0.56	1.7507	0.24320	0.57121
0.22	1.2461	0.09554	0.80252	0.57	1.7683	0.24755	0.56553
0.23	1.2586	0.09989	0.79453	0.58	1.7860	0.25189	0.55990
0.24	1.2712	0.10423	0.78663	0.59	1.8040	0.25623	0.55433
0.25	1.2840	0.10857	0.77880	0.60	1.8221	0.26058	0.54881
0.26	1.2969	0.11292	0.77105	0.61	1.8404	0.26492	0.54335
0.27	1.3100	0.11726	0.76338	0.62	1.8589	0.26926	0.53794
0.28	1.3231	0.12160	0.75578	0.63	1.8776	0.27361	0.53259
0.29	1.3364	0.12595	0.74826	0.64	1.8965	0.27795	0.52729
0.30	1.3499	0.13029	0.74082	0.65	1.9155	0.28229	0.52205
0.31	1.3634	0.13463	0.73345	0.66	1.9348	0.28664	0.51685
0.32	1.3771	0.13897	0.72615	0.67	1.9542	0.29098	0.51171
0.33	1.3910	0.14332	0.71892	0.68	1.9739	0.29532	0.50662
0.34	1.4049	0.14766	0.71177	0.69	1.9937	0.29966	0.50158

Table 1-9. Values and Logarithms of Exponential Functions (Continued)

x	e^x		e^{-x} value	x	e^x		e^{-x} value
	Value	$\log_{10}$			Value	$\log_{10}$	
0.70	2.0138	0.30401	0.49659	1.05	2.8577	0.45601	0.34994
0.71	2.0340	0.30835	0.49164	1.06	2.8864	0.46035	0.34646
0.72	2.0544	0.31269	0.48675	1.07	2.9154	0.46470	0.34301
0.73	2.0751	0.31703	0.48191	1.08	2.9447	0.46904	0.33960
0.74	2.0959	0.32138	0.47711	1.09	2.9743	0.47338	0.33622
0.75	2.1170	0.32572	0.47237	1.10	3.0042	0.47772	0.33287
0.76	2.1383	0.33006	0.46767	1.11	3.0344	0.48207	0.32956
0.77	2.1598	0.33441	0.46301	1.12	3.0649	0.48641	0.32628
0.78	2.1815	0.33875	0.45841	1.13	3.0957	0.49075	0.32303
0.79	2.2034	0.34309	0.45384	1.14	3.1268	0.49510	0.31982
0.80	2.2255	0.34744	0.44933	1.15	3.1582	0.49944	0.31664
0.81	2.2479	0.35178	0.44486	1.16	3.1899	0.50378	0.31349
0.82	2.2705	0.35612	0.44043	1.17	3.2220	0.50812	0.31037
0.83	2.2933	0.36046	0.43605	1.18	3.2544	0.51247	0.30728
0.84	2.3164	0.36481	0.43171	1.19	3.2871	0.51681	0.30422
0.85	2.3396	0.36915	0.42741	1.20	3.3201	0.52115	0.30119
0.86	2.3632	0.37349	0.42316	1.21	3.3535	0.52550	0.29820
0.87	2.3869	0.37784	0.41895	1.22	3.3872	0.52984	0.29523
0.88	2.4109	0.38218	0.41478	1.23	3.4212	0.53418	0.29229
0.89	2.4351	0.38652	0.41066	1.24	3.4556	0.53853	0.28938
0.90	2.4596	0.39087	0.40657	1.25	3.4903	0.54287	0.28650
0.91	2.4843	0.39521	0.40252	1.26	3.5254	0.54721	0.28365
0.92	2.5093	0.39955	0.39852	1.27	3.5609	0.55155	0.28083
0.93	2.5345	0.40389	0.39455	1.28	3.5966	0.55590	0.27804
0.94	2.5600	0.40824	0.39063	1.29	3.6328	0.56024	0.27527
0.95	2.5857	0.41258	0.38674	1.30	3.6693	0.56458	0.27253
0.96	2.6117	0.41692	0.38289	1.31	3.7062	0.56893	0.26982
0.97	2.6379	0.42127	0.37908	1.32	3.7434	0.57327	0.26714
0.98	2.6645	0.42561	0.37531	1.33	3.7810	0.57761	0.26448
0.99	2.6912	0.42995	0.37158	1.34	3.8190	0.58195	0.26185
1.00	2.7183	0.43429	0.36788	1.35	3.8574	0.58630	0.25924
1.01	2.7456	0.43864	0.36422	1.36	3.8962	0.59064	0.25666
1.02	2.7732	0.44298	0.36060	1.37	3.9354	0.59498	0.25411
1.03	2.8011	0.44732	0.35701	1.38	3.9749	0.59933	0.25158
1.04	2.8292	0.45167	0.35345	1.39	4.0149	0.60367	0.24908

Table 1-9. Values and Logarithms of Exponential Functions (Continued)

z	e^z		e^{-z} value	z	e^z		e^{-z} value
	Value	log ₁₀			Value	log ₁₀	
1.40	4.0552	0.60801	0.24660	1.75	5.7546	0.76002	0.17377
1.41	4.0960	0.61236	0.24414	1.76	5.8124	0.76436	0.17204
1.42	4.1371	0.61670	0.24171	1.77	5.8709	0.76870	0.17033
1.43	4.1787	0.62104	0.23931	1.78	5.9299	0.77304	0.16864
1.44	4.2207	0.62538	0.23693	1.79	5.9895	0.77739	0.16696
1.45	4.2631	0.62973	0.23457	1.80	6.0496	0.78173	0.16530
1.46	4.3060	0.63407	0.23224	1.81	6.1104	0.78607	0.16365
1.47	4.3492	0.63841	0.22993	1.82	6.1719	0.79042	0.16203
1.48	4.3929	0.64276	0.22764	1.83	6.2339	0.79476	0.16041
1.49	4.4371	0.64710	0.22537	1.84	6.2965	0.79910	0.15882
1.50	4.4817	0.65144	0.22313	1.85	6.3598	0.80344	0.15724
1.51	4.5267	0.65578	0.22091	1.86	6.4237	0.80779	0.15567
1.52	4.5722	0.66013	0.21871	1.87	6.4883	0.81213	0.15412
1.53	4.6182	0.66447	0.21654	1.88	6.5535	0.81647	0.15259
1.54	4.6646	0.66881	0.21438	1.89	6.6194	0.82082	0.15107
1.55	4.7115	0.67316	0.21225	1.90	6.6859	0.82516	0.14957
1.56	4.7588	0.67750	0.21014	1.91	6.7531	0.82950	0.14808
1.57	4.8066	0.68184	0.20805	1.92	6.8210	0.83385	0.14661
1.58	4.8550	0.68619	0.20598	1.93	6.8895	0.83819	0.14515
1.59	4.9037	0.69053	0.20393	1.94	6.9588	0.84253	0.14370
1.60	4.9530	0.69487	0.20190	1.95	7.0287	0.84687	0.14227
1.61	5.0028	0.69921	0.19989	1.96	7.0993	0.85122	0.14086
1.62	5.0531	0.70356	0.19790	1.97	7.1707	0.85556	0.13946
1.63	5.1039	0.70790	0.19593	1.98	7.2427	0.85990	0.13807
1.64	5.1552	0.71224	0.19398	1.99	7.3155	0.86425	0.13670
1.65	5.2070	0.71659	0.19205	2.00	7.3891	0.86859	0.13534
1.66	5.2593	0.72093	0.19014	2.01	7.4633	0.87293	0.13399
1.67	5.3122	0.72527	0.18825	2.02	7.5383	0.87727	0.13266
1.68	5.3656	0.72961	0.18637	2.03	7.6141	0.88162	0.13134
1.69	5.4195	0.73396	0.18452	2.04	7.6906	0.88596	0.13003
1.70	5.4739	0.73830	0.18268	2.05	7.7679	0.89030	0.12873
1.71	5.5290	0.74264	0.18087	2.06	7.8460	0.89465	0.12745
1.72	5.5845	0.74699	0.17907	2.07	7.9248	0.89899	0.12619
1.73	5.6407	0.75133	0.17728	2.08	8.0045	0.90333	0.12493
1.74	5.6973	0.75567	0.17552	2.09	8.0849	0.90768	0.12369

Table 1-9. Values and Logarithms of Exponential Functions (Continued)

x	e^x		e^{-x} value	x	e^x		e^{-x} value
	Value	log ₁₀			Value	log ₁₀	
2.10	8.1662	0.91202	0.12246	2.45	11.588	1.06402	0.08629
2.11	8.2482	0.91636	0.12124	2.46	11.705	1.06836	0.08543
2.12	8.3311	0.92070	0.12003	2.47	11.822	1.07271	0.08458
2.13	8.4149	0.92505	0.11884	2.48	11.941	1.07705	0.08374
2.14	8.4994	0.92939	0.11765	2.49	12.061	1.08139	0.08291
2.15	8.5849	0.93373	0.11648	2.50	12.182	1.08574	0.08208
2.16	8.6711	0.93808	0.11533	2.51	12.305	1.09008	0.08127
2.17	8.7583	0.94242	0.11418	2.52	12.429	1.09442	0.08046
2.18	8.8463	0.94676	0.11304	2.53	12.554	1.09877	0.07966
2.19	8.9352	0.95110	0.11192	2.54	12.680	1.10311	0.07887
2.20	9.0250	0.95545	0.11080	2.55	12.807	1.10745	0.07808
2.21	9.1157	0.95979	0.10970	2.56	12.936	1.11179	0.07730
2.22	9.2073	0.96413	0.10861	2.57	13.066	1.11614	0.07654
2.23	9.2999	0.96848	0.10753	2.58	13.197	1.12048	0.07577
2.24	9.3933	0.97282	0.10646	2.59	13.330	1.12482	0.07502
2.25	9.4877	0.97716	0.10540	2.60	13.464	1.12917	0.07427
2.26	9.5831	0.98151	0.10435	2.61	13.599	1.13351	0.07353
2.27	9.6794	0.98585	0.10331	2.62	13.736	1.13785	0.07280
2.28	9.7767	0.99019	0.10228	2.63	13.874	1.14219	0.07208
2.29	9.8749	0.99453	0.10127	2.64	14.013	1.14654	0.07136
2.30	9.9742	0.99888	0.10026	2.65	14.154	1.15088	0.07065
2.31	10.074	1.00322	0.09926	2.66	14.296	1.15522	0.06995
2.32	10.176	1.00756	0.09827	2.67	14.440	1.15957	0.06925
2.33	10.278	1.01191	0.09730	2.68	14.585	1.16391	0.06856
2.34	10.381	1.01625	0.09633	2.69	14.732	1.16825	0.06788
2.35	10.486	1.02059	0.09537	2.70	14.880	1.17260	0.06721
2.36	10.591	1.02493	0.09442	2.71	15.029	1.17694	0.06654
2.37	10.697	1.02928	0.09348	2.72	15.180	1.18128	0.06587
2.38	10.805	1.03362	0.09255	2.73	15.333	1.18562	0.06522
2.39	10.913	1.03796	0.09163	2.74	15.487	1.18997	0.06457
2.40	11.023	1.04231	0.09072	2.75	15.643	1.19431	0.06393
2.41	11.134	1.04665	0.08982	2.76	15.800	1.19865	0.06329
2.42	11.246	1.05099	0.08892	2.77	15.959	1.20300	0.06266
2.43	11.359	1.05534	0.08804	2.78	16.119	1.20734	0.06204
2.44	11.473	1.05968	0.08716	2.79	16.281	1.21168	0.06142

Table 1-9. Values and Logarithms of Exponential Functions (Continued)

x	e^x		e^{-x} value	x	e^x		e^{-x} value
	Value	$\log_{10}$			Value	$\log_{10}$	
2.80	16.445	1.21602	0.06081	2.90	18.174	1.25945	0.05502
2.81	16.610	1.22037	0.06020	2.91	18.357	1.26380	0.05448
2.82	16.777	1.22471	0.05961	2.92	18.541	1.26814	0.05393
2.83	16.945	1.22905	0.05901	2.93	18.728	1.27248	0.05340
2.84	17.116	1.23340	0.05843	2.94	18.916	1.27683	0.05287
2.85	17.288	1.23774	0.05784	2.95	19.106	1.28117	0.05234
2.86	17.462	1.24208	0.05727	2.96	19.298	1.28551	0.05182
2.87	17.637	1.24643	0.05670	2.97	19.492	1.28985	0.05130
2.88	17.814	1.25077	0.05613	2.98	19.688	1.29420	0.05079
2.89	17.993	1.25511	0.05558	2.99	19.886	1.29854	0.05029
				3.00	20.086	1.30288	0.04979

Table 1-10. Air-density Correction Factors for Air-ionization Measurements *

(Air density relative to dry air at 760 mm and 25°C)

Pres- sure, mm	Temperature, °C														
	-40	-30	-20	-10	-5	0	5	10	15	20	25	30	35	40	45
660	1.111	1.065	1.023	0.984	0.966	0.948	0.931	0.914	0.899	0.883	0.868	0.854	0.840	0.827	0.814
670	1.128	1.081	1.038	0.999	0.980	0.962	0.945	0.928	0.912	0.897	0.882	0.867	0.853	0.839	0.826
680	1.144	1.097	1.054	1.014	0.995	0.977	0.959	0.942	0.926	0.910	0.895	0.880	0.866	0.852	0.838
690	1.161	1.113	1.069	1.029	1.010	0.991	0.973	0.956	0.939	0.923	0.908	0.893	0.878	0.864	0.851
700	1.178	1.130	1.085	1.044	1.024	1.005	0.987	0.970	0.953	0.937	0.921	0.906	0.891	0.877	0.863
710	1.195	1.146	1.100	1.059	1.039	1.020	1.001	0.984	0.967	0.950	0.934	0.919	0.904	0.889	0.875
720	1.212	1.162	1.116	1.074	1.053	1.034	1.016	0.998	0.980	0.964	0.947	0.932	0.917	0.902	0.888
730	1.229	1.178	1.131	1.088	1.068	1.048	1.030	1.011	0.994	0.977	0.961	0.945	0.929	0.914	0.900
740	1.245	1.194	1.147	1.103	1.083	1.063	1.044	1.025	1.007	0.990	0.974	0.958	0.942	0.927	0.912
750	1.262	1.210	1.162	1.118	1.097	1.077	1.058	1.039	1.021	1.004	0.987	0.971	0.955	0.940	0.925
760	1.279	1.226	1.178	1.133	1.112	1.092	1.072	1.053	1.035	1.017	1.000	0.984	0.968	0.952	0.937
770	1.296	1.242	1.193	1.148	1.127	1.105	1.086	1.067	1.048	1.030	1.013	0.996	0.980	0.965	0.949
780	1.313	1.259	1.209	1.163	1.140	1.120	1.100	1.081	1.062	1.044	1.026	1.009	0.993	0.977	0.962
790	1.330	1.275	1.224	1.178	1.156	1.135	1.114	1.095	1.076	1.057	1.039	1.022	1.006	0.990	0.974
800	1.346	1.291	1.240	1.194	1.170	1.149	1.128	1.108	1.089	1.071	1.053	1.035	1.018	1.002	0.986

* Adapted from Knowlton, A. E. (ed.), "Standard Handbook for Electrical Engineers," 7th ed., McGraw-Hill, 1941.

Table 1-11. Properties of the Earth's Atmosphere *

Altitude, km above sea level	Pressure, mm Hg	Density, g/m ³	Temp, °K (N ₂ , O ₂) M = 29	Temp, °K (N ₂ , O) M = 24	Velocity of sound, m/sec	Mean free path, cm (N ₂)
0	760	1,220	290		345	6.5×10^{-6}
10	210	425	230		310	1.9×10^{-6}
20	42	92	210		295	8.6×10^{-6}
30	9.5	19	235		315	4.2×10^{-4}
40	2.4	4.3	260		325	1.8×10^{-3}
50	7.5×10^{-1}	1.3	270		330	6.1×10^{-3}
60	2.1×10^{-1}	3.8×10^{-1}	260		325	2.1×10^{-2}
70	5.4×10^{-2}	1.2×10^{-1}	210		295	6.6×10^{-1}
80	1.0×10^{-2}	2.5×10^{-2}	190		280	3.2×10^{-1}
90	1.9×10^{-3}	4.0×10^{-3}	210		295	2.0
100	4.2×10^{-4}	8×10^{-4}	240		315	10.0
110	1.2×10^{-4}	2.0×10^{-4}	270	220	330	40
120	3.5×10^{-5}	5.0×10^{-5}	330	270	370	1.5×10^2
130	1.5×10^{-5}	2.0×10^{-5}	390	320	400	4×10^2
140	7×10^{-6}	7×10^{-6}	450	370	430	1×10^3
150	3×10^{-6}	3.0×10^{-6}	510	420	460	2.5×10^3
160	2×10^{-6}	1.5×10^{-6}	570	470	480	5×10^3

The average atmosphere up to 160 km based on pressure and density data obtained on rocket flights above White Sands, N.M.

* Havens, Koll, and LaGow, *J. Geophys. Research*, March, 1952.

Table 1-12. Weather Data—U.S. Weather Bureau *

State and city	Average temperature, Oct. 1 to May 1	Highest temperature ever occurred, °F	Lowest temperature ever occurred, °F	Average wind velocity December, January, February, mph	Direction of prevailing wind December, January, February
Alabama:					
Mobile.....	59.2	103	— 1	9.9	N
Birmingham.....	54.1	107	—10	8.0	N
Arizona:					
Phoenix.....	60.0	118	16	5.4	SW
Flagstaff.....	34.9	99	—30	6.7	SW
Arkansas:					
Fort Smith.....	50.9	113	—15	8.3	E
Little Rock.....	51.8	110	—12	8.3	NW
California:					
San Francisco.....	54.6	101	27	7.5	N
Los Angeles.....	59.3	109	28	6.4	NE
Colorado:					
Denver.....	39.3	105	—29	7.5	S
Grand Junction....	39.7	105	—21	4.4	SE
Connecticut:					
New Haven.....	38.6	101	—15	9.4	N
District of Columbia:					
Washington.....	44.0	106	—15	7.7	NW
Florida:					
Jacksonville.....	62.2	104	10	9.0	NE
Georgia:					
Atlanta.....	53.3	103	— 8	11.7	NW
Savannah.....	58.8	105	8	9.5	NW
Idaho:					
Lewiston.....	42.6	...	—23	4.7	E
Pocatello.....	36.5	105	—28	8.9	SW
Illinois:					
Chicago.....	36.9	105	—23	12.0	SW
Springfield.....	40.5	110	—24	11.9	NW
Indiana:					
Indianapolis.....	40.5	106	—25	11.3	S
Evansville.....	45.2	108	—16	9.6	SW
Iowa:					
Dubuque.....	34.3	110	—32	7.1	NW
Sioux City.....	34.0	111	—35	11.5	NW
Kansas:					
Concordia.....	39.8	114	—25	7.7	S
Dodge City.....	41.9	109	—26	10.6	NW

* By F. W. Reichelderfer, Chief Weather Bureau, U.S. Department of Commerce. From Perry, J. H. (ed.), "Chemical Engineers' Handbook," 3d ed., McGraw-Hill, 1950

Table 1-12. Weather Data—U.S. Weather Bureau (Continued)

State and city	Average tempera- ture, Oct. 1 to May 1	Highest tempera- ture ever occurred, °F	Lowest tempera- ture ever occurred, °F	Average wind velocity December, January, February, mph	Direction of prevailing wind December, January, February
Kentucky:					
Louisville.....	45.2	107	-20	9.9	SW
Louisiana:					
New Orleans.....	61.9	102	7	8.6	NE
Shreveport.....	56.7	110	-5	8.8	SE
Maine:					
Eastport.....	31.7	93	-23	12.6	NW
Portland.....	34.0	103	-21	10.4	NW
Maryland:					
Baltimore.....	44.2	107	-7	8.2	SW
Massachusetts:					
Boston.....	38.4	104	-18	12.4	W
Michigan:					
Alpena.....	30.0	104	-28	11.0	W
Detroit.....	35.9	105	-24	12.0	SW
Marquette.....	31.2	108	-27	10.6	NW
Minnesota:					
Duluth.....	24.3	106	-41	13.4	NW
Minneapolis.....	29.9	108	-34	11.3	NW
Mississippi:					
Vicksburg.....	56.9	104	-1	8.3	SE
Missouri:					
St. Joseph.....	41.2	110	-24	9.3	NW
Springfield.....	44.5	106	-29	10.9	SE
Montana:					
Billings.....	35.2	112	-49	12.4	SW
Havre.....	28.5	108	-57	9.4	SW
Nebraska:					
Lincoln.....	37.9	115	-29	10.6	S
North Platte.....	36.3	109	-35	7.9	W
Nevada:					
Tonopah.....	39.4	98	-15	9.9	SE
Winnemucca.....	38.0	108	-36	8.1	NE
New Hampshire:					
Concord.....	31.2	102	-37	6.2	NW
New Jersey:					
Atlantic City.....	42.2	104	-9	15.8	NW
New Mexico:					
Santa Fe.....	38.5	97	-13	7.0	N

Table 1-12. Weather Data—U.S. Weather Bureau (Continued)

State and city	Average temperature, Oct. 1 to May 1	Highest temperature ever occurred, °F	Lowest temperature ever occurred, °F	Average wind velocity December, January, February, mph	Direction of prevailing wind December, January, February
New York:					
Albany.....	35.3	104	-24	10.5	W
Buffalo.....	35.0	97	-20	17.1	W
New York.....	40.9	102	-14	16.9	NW
North Carolina:					
Raleigh.....	50.2	104	- 2	7.9	SW
Wilmington.....	54.5	103	5	9.4	N
North Dakota:					
Bismarek.....	25.4	114	-45	9.1	NW
Devil's Lake.....	21.9	112	-46	10.1	NW
Ohio:					
Cleveland.....	36.1	103	-17	14.7	SW
Columbus.....	40.3	106	-20	11.6	SW
Oklahoma:					
Oklahoma City....	48.6	113	-17	11.5	S
Oregon:					
Baker.....	35.6	104	-25	6.6	SE
Portland.....	46.1	107	- 2	7.3	S
Pennsylvania:					
Philadelphia.....	42.7	106	-11	11.0	NW
Pittsburgh.....	40.7	103	-20	11.7	W
Rhode Island:					
Providence.....	38.3	100	-17	12.2	NW
South Carolina:					
Charleston.....	57.5	104	7	10.5	N
Columbia.....	54.3	106	- 2	8.1	NE
South Dakota:					
Huron.....	29.3	111	-43	10.7	NW
Rapid City.....	34.0	106	-34	7.9	W
Tennessee:					
Knoxville.....	48.1	104	-16	7.2	SW
Memphis.....	51.3	106	- 9	9.3	NW
Texas:					
El Paso.....	53.8	106	- 6	9.0	NW
Fort Worth.....	55.5	112	- 8	10.4	NW
San Antonio.....	60.9	107	4	8.3	NE
Utah:					
Modena.....	36.7	101	-32	8.4	W
Salt Lake City....	32.7	105	-30	7.9	SE

Table 1-12. Weather Data—U.S. Weather Bureau (Continued)

State and city	Average temperature, Oct. 1 to May 1	Highest temperature ever occurred, °F	Lowest temperature ever occurred, °F	Average wind velocity December, January, February, mph	Direction of prevailing wind December, January, February
Vermont:					
Burlington.....	31.2	101	-29	11.7	S
Virginia:					
Norfolk.....	49.6	105	2	12.1	N
Lynchburg.....	46.6	106	-7	8.1	NW
Richmond.....	47.5	107	-3	8.1	SW
Washington:					
Seattle.....	45.8	100	3	9.7	SE
Spokane.....	37.9	108	-30	6.2	SW
West Virginia:					
Elkins.....	39.7	99	-28	6.1	W
Parkersburg.....	42.7	106	-27	7.2	SW
Wisconsin:					
Green Bay.....	30.6	104	-36	10.5	SW
LaCrosse.....	32.0	108	-43	2.2	NW
Milwaukee.....	33.7	105	-25	12.1	W
Wyoming:					
Sheridan.....	31.0	106	-41	5.4	NW
Lander.....	30.3	102	-40	3.9	SW

Table 1-13. Densities of Common Metals
(20 to 25°C)

Metal	Symbol	Density	Metal	Symbol	Density
Aluminum.....	Al	2.70	Nickel.....	Ni	8.9
Antimony.....	Sb	6.685	Osmium.....	Os	22.48
Arsenic.....	As	5.7	Palladium.....	Pd	12.1
Barium.....	Ba	3.5	Platinum.....	Pt	21.4
Beryllium.....	Be	1.8	Potassium.....	K	0.86
Bismuth.....	Bi	9.8	Praseodymium...	Pr	6.5
Cadmium.....	Cd	8.6	Rhenium.....	Re	20.53
Calcium.....	Ca	1.55	Rhodium.....	Rh	12.5
Cerium.....	Ce	6.90	Rubidium.....	Rb	1.53
Cesium.....	Cs	1.9	Ruthenium.....	Ru	12.2
Chromium.....	Cr	7.1	Selenium.....	Se	4.8
Cobalt.....	Co	8.9	Silver.....	Ag	10.5
Copper.....	Cu	8.93	Sodium.....	Na	0.97
Gallium.....	Ga	5.91	Strontium.....	Sr	2.6
Germanium.....	Ge	5.36	Tantalum.....	Ta	16.6
Gold.....	Au	19.3	Tellurium.....	Te	6.24
Indium.....	In	7.3	Thallium.....	Tl	11.85
Iridium.....	Ir	22.4	Thorium.....	Th	11.4
Iron.....	Fe	7.86	Tin (white).....	Sn	7.31
Lanthanum.....	La	6.15	Tin (gray).....	Sn	5.75
Lead.....	Pb	11.34	Titanium.....	Ti	4.5
Lithium.....	Li	0.53	Uranium.....	U	18.7
Magnesium.....	Mg	1.74	Vanadium.....	V	5.96
Manganese.....	Mn	7.2	Wolfram.....	W	19.3
Mercury.....	Hg	13.546 (liq)	Yttrium.....	Y	5.51
Molybdenum.....	Mo	10.2	Zinc.....	Zn	7.14
Neodymium.....	Nd	6.9	Zirconium.....	Zr	6.4

Table 1-14. Sheet-lead Thicknesses (As Used in Shielding)

Millimeters	Inches	Pounds per square foot
0.8	$\frac{1}{32}$	2
1.0		$2\frac{1}{2}$
1.2	$\frac{3}{64}$	3
1.5-1.6	$\frac{1}{16}$	4
1.9-2.0	$\frac{5}{64}$	5
2.4	$\frac{3}{32}$	6
3.2	$\frac{1}{8}$	8
4.0	$\frac{5}{32}$	10
4.8	$\frac{3}{16}$	12
5.6	$\frac{7}{32}$	14
6.4	$\frac{1}{4}$	16
7.4	0.29	18
8.6	0.34	20

Heavier sheets are usually rolled to order.

Table 1-15. Relation between Thickness of Ordinary Concrete and Thickness of Lead for Radium and Cobalt-60 Gamma Rays (Broad-beam Conditions)

Thickness of concrete, in.		Thickness of lead, mm	
A	B	Radium gamma-rays	Cobalt-60 gamma-rays
2	$17\frac{1}{8}$	5	7
4	$3\frac{3}{4}$	12	15
6	$5\frac{5}{8}$	19	24
8	$7\frac{1}{2}$	27	32
10	$9\frac{3}{8}$	35	41
12	$11\frac{1}{4}$	43	50
14	$13\frac{1}{8}$	52	59
16	15	60	69
18	$16\frac{7}{8}$	69	79
20	$18\frac{3}{4}$	77	90
22	$20\frac{5}{8}$	86	100
24	$22\frac{1}{2}$	94	110

A. Concrete density of 2.2 g/cm³ as used in England.

B. Concrete density of 2.35 g/cm³ as used in United States of America.

Table 1-16. Relation between Thickness of Ordinary Concrete and Lead Equivalent at Various X-ray Qualities (Broad-beam Conditions)

Thickness of concrete, in.		Lead equivalent, mm, for X-rays excited at the following kilovoltages										
		50	75	100	150	200	250	300	400	500	1,000	2,000
A	B											
2	17 ⁷ / ₈	0.4	0.5	0.6	0.5	0.5	0.6	0.8	1.1	1.6	4.0	6
4	33 ³ / ₄	0.9	1.2	1.4	1.2	1.2	1.7	2.2	3.0	3.9	8.6	13
6	55 ⁵ / ₈	1.4	2.0	2.4	1.9	2.1	3.0	3.8	5.4	7.1	13	22
8	71 ¹ / ₂	2.0	2.8	3.4	2.7	2.9	4.4	5.8	8.5	11	21	31
10	93 ³ / ₈	2.5	3.6	4.4	3.4	3.8	5.8	7.9	11	15	29	40
12	111 ¹ / ₄	3.1	4.3	5.4	4.2	4.7	7.3	10	14	19	37	49
14	131 ¹ / ₈	...	...	...	5.1	5.6	8.6	12	18	24	45	58
16	15	...	...	...	...	...	...	...	21	28	54	67
18	167 ⁷ / ₈	...	...	...	...	...	...	...	24	33	62	76
20	183 ³ / ₄	...	...	...	...	...	...	...	...	37	71	85
24	221 ¹ / ₂	...	...	...	...	...	...	...	...	46	88	103
30	28	...	...	...	...	...	...	...	...	60	112	130
36	335 ⁵ / ₈	...	...	...	...	...	...	...	...	...	138	159

A. Concrete density of 2.2 g/cm³ as used in England.B. Concrete density of 2.35 g/cm³ as used in United States of America.**Table 1-17. Relation between Thickness of Brick and Lead Equivalent at Various X-ray Qualities (Broad-beam Conditions)**

Thickness of brick, in.		Lead equivalent, mm, for X-rays excited at the following kilovoltages							
		50	75	100	150	200	250	300	400
A	B								
4 $\frac{1}{2}$	3 $\frac{3}{4}$	0.7	0.9	1.0	0.9	0.9	1.2	1.3	1.5
9	7 $\frac{1}{2}$	1.6	2.0	2.4	2.0	2.0	2.8	3.7	5.6
13 $\frac{1}{2}$	11 $\frac{3}{8}$	...	3.2	3.7	3.1	3.1	4.9	6.8	10.3
18	15 $\frac{1}{8}$	...	...	...	4.3	4.3	7.2	10	15.2

A. British yellow stock brick (density 1.6 g/cm³).B. American common brick (density 1.9 g/cm³).

Table 1-18. The Standard Man (Conventional): Masses of Organs in Human Body

Organ	Mass, g
Total body.....	70,000
Muscle.....	30,000
Skin and subcutaneous tissue.....	6,100
Skin only.....	2,000
Fat.....	10,000
Skeleton:	
Without bone marrow.....	7,000
Red marrow.....	1,500
Yellow marrow.....	1,500
Blood.....	5,400
Gastrointestinal tract.....	2,000
Contents of gastrointestinal tract:	
Stomach.....	250
Small intestine.....	1,100
Upper large intestine.....	135
Lower large intestine.....	150
Liver.....	1,700
Brain.....	1,500
Lungs (2).....	1,000
Lymphoid tissue.....	700
Kidneys (2).....	300
Heart.....	300
Spleen.....	150
Urinary bladder.....	150
Pancreas.....	70
Salivary glands (6).....	50
Testes (2).....	40
Spinal cord.....	30
Eyes (2).....	30
Thyroid gland.....	20
Teeth.....	20
Prostate gland.....	20
Adrenal glands or suprarenal (2).....	20
Thymus.....	10
Miscellaneous (blood vessels, cartilage, nerves, etc.).....	390

In calculations of the maximum permissible body burdens and the maximum permissible concentrations in air and water of various radioactive isotopes, data are required concerning the masses of the body organs and the rates of inhalation and ingestion. The data accepted at the present time for the conventional "standard man" are given in Tables 1-18 to 1-20.

Table 1-19. The Standard Man (Conventional): Daily Rates of Ingestion and Inhalation

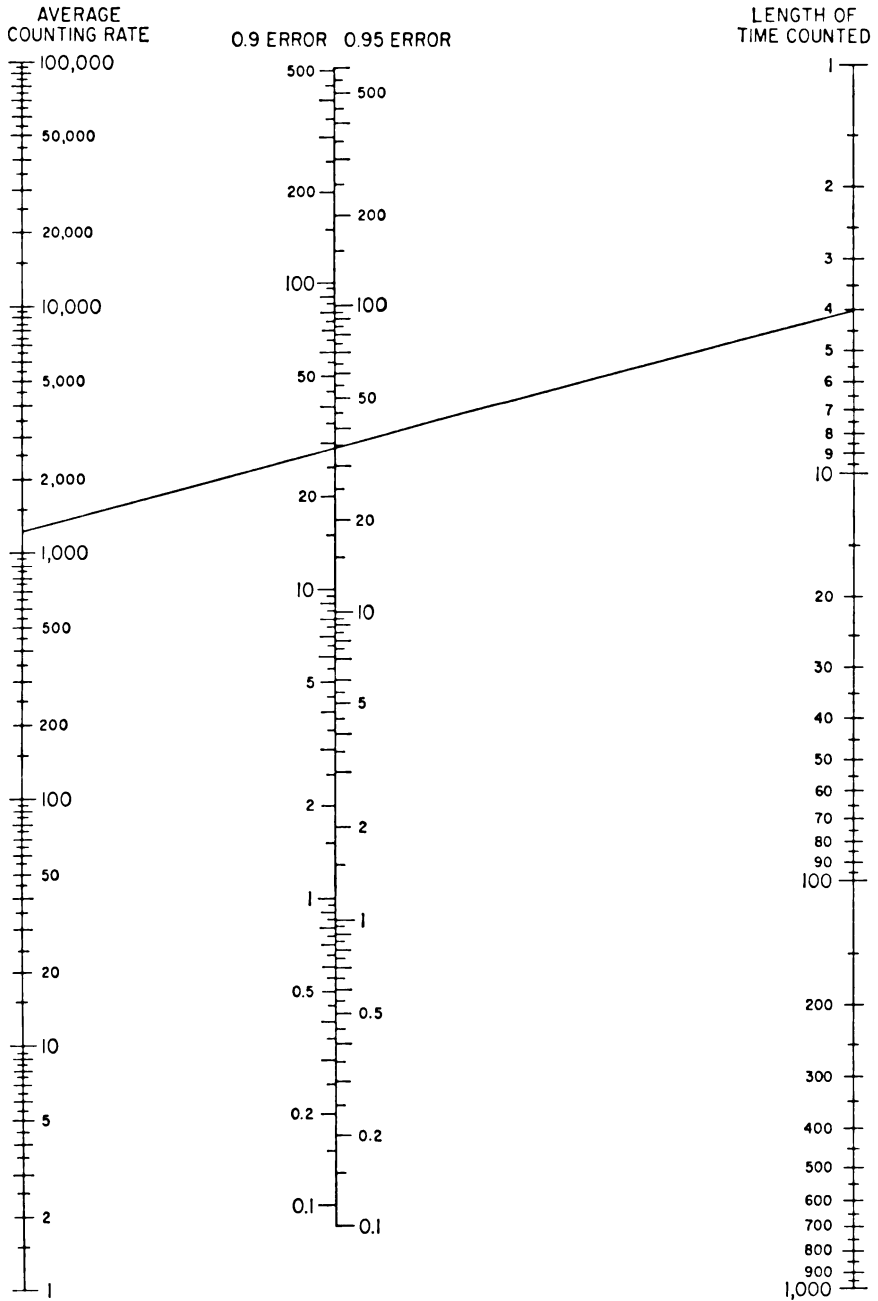
	<i>cm</i> ³
Water intake in food	700
Water intake in fluids	1,500
Water of oxidation	300
Total water consumption	2,500
Air inhaled during 8-hr working day	10 ⁷
Air inhaled during 16 hr not at work	10 ⁷
Total inhaled	2 × 10 ⁷

As regards the distribution of inhaled particulates in the respiratory tract, values are available for certain isotopes. Where data are lacking, the convention given in Table 1-20 is adopted.

Table 1-20. The Standard Man: Particulates in Respiratory Tract

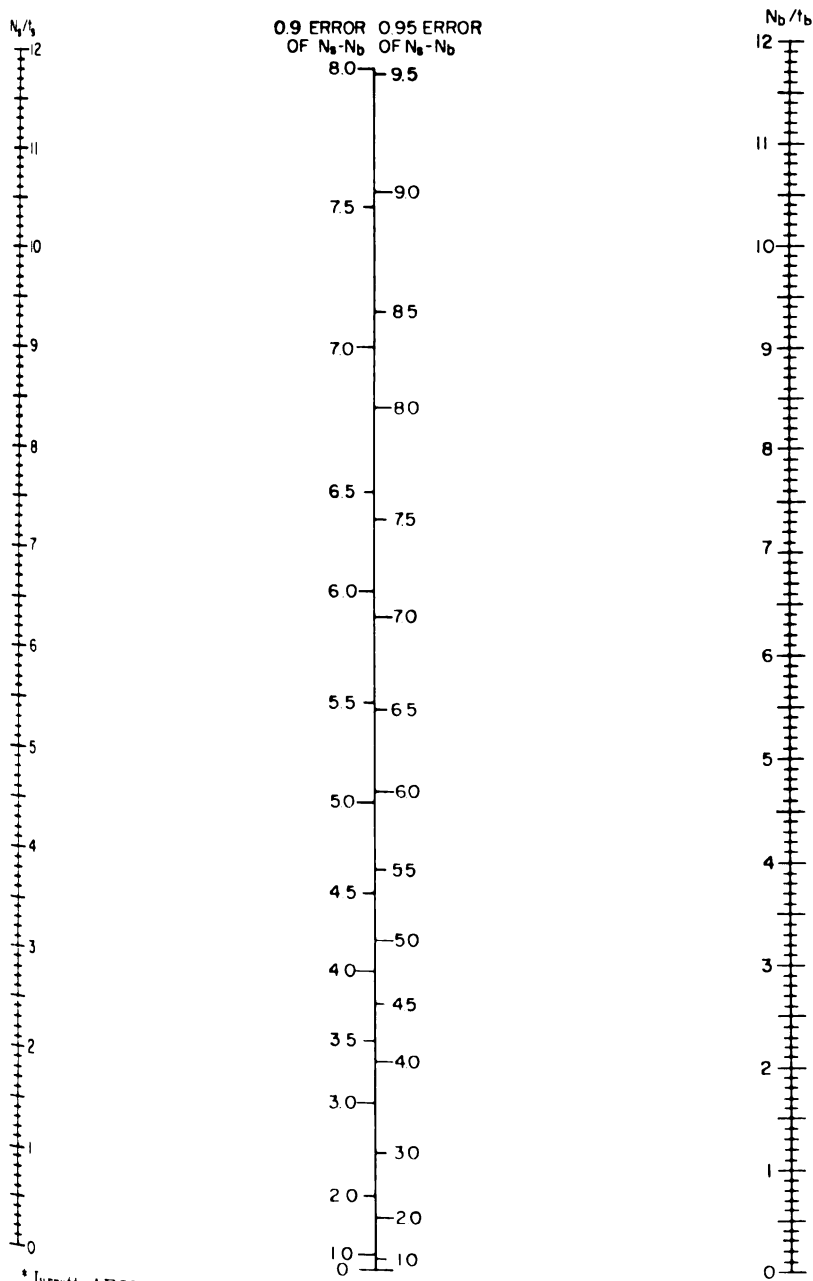
Distribution	Readily soluble compounds, %	Other compounds, %
Exhaled	25	25
Deposited in upper respiratory passages and ultimately swallowed	50	50
Deposited in lungs	25 (taken up in body fluids)	25 (12.5% eliminated from lungs and swallowed in first 24 hours; remaining 12.5% retained in the lungs with a half-life of 120 days and then taken up into the body fluids)

Table 1-21a. Count-rate Nomograph—90 and 95 Per Cent Error—High Counting Rates *



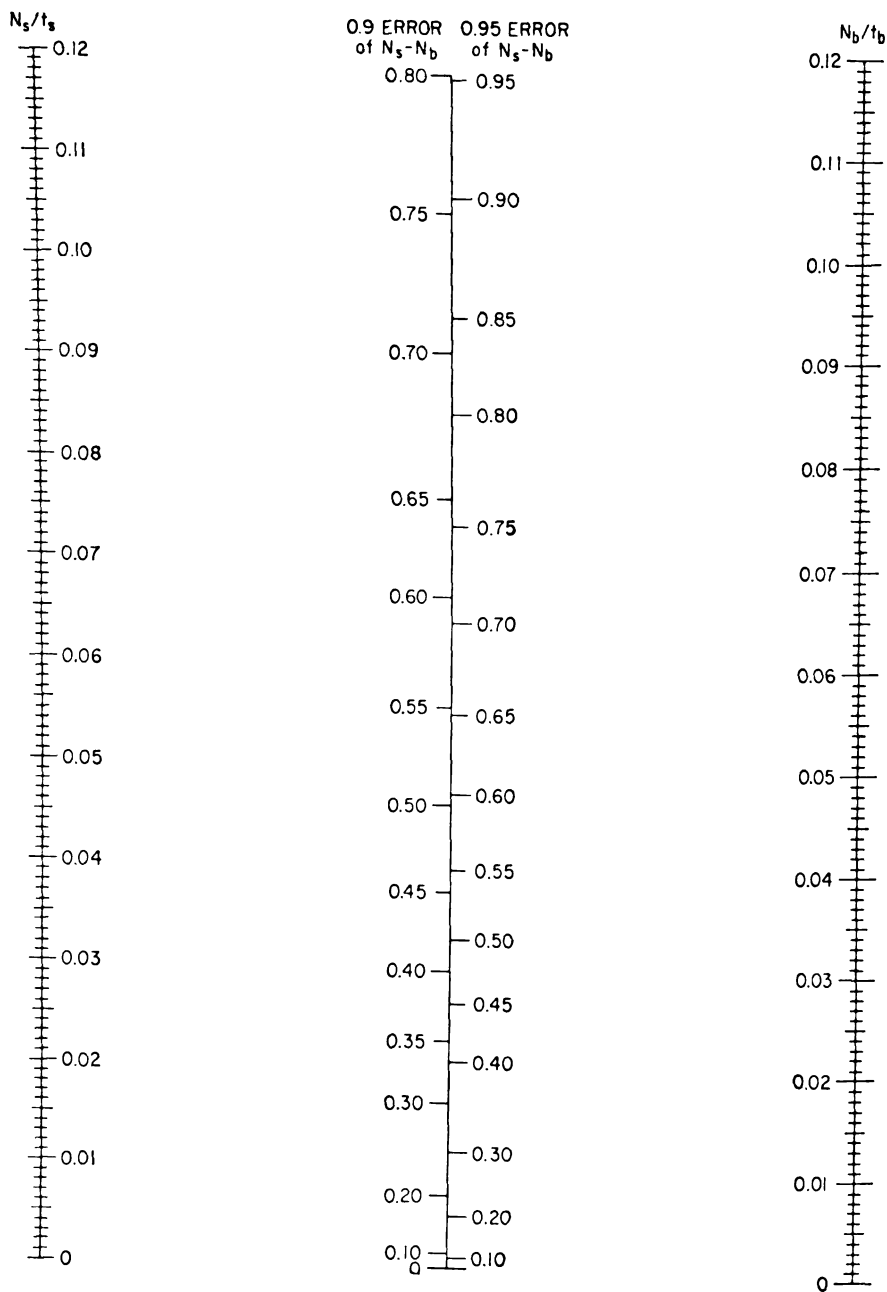
* Jarrett, AECU-262, MonP-126. See instructions for use on page 1-43.

Table 1-21b. Count-rate Nomograph—90 and 95 Per Cent Error—Low Counting Rates *



* Jarrett, AECU-262, MonP-126. See instructions for use on page 1-43.

Table 1-21b. Count-rate Nomograph—90 and 95 Per Cent Error—Low Counting Rates * (Continued)



* Jarrett, AECU-262, MonP-126.

Instructions for Use of Table 1-21a

Instructions for Use: Draw a straight line from a point on the left scale corresponding to the counting rate of the sample through the point on the right scale corresponding to the length of time the sample was counted. The point where this line crosses the center scale corresponds to the 0.9 error and the 0.95 error of the determination.

Instructions for Use of Table 1-21b

Instructions for Use: Draw a straight line from a point on the left scale that corresponds to the quotient N_s/t_s through the point on the right scale that corresponds to the quotient N_b/t_b . The point where this line crosses the center scale will correspond to the 0.9 and the 0.95 error of the determination $N_s - N_b$.

Explanation of Symbols:

N_s = counting rate of sample including background in counts per minute

t_s = number of minutes sample was counted

N_b = counting rate of background in counts per minute

t_b = number of minutes background was counted

Table 1-22. Multiple Half-lives and Half-value Layers *

Number of Half-lives, * n	Dose Rate or Count Rate after n Half-lives, * I_n
0	1.00
1	0.50
2	0.25
3.3	10^{-1} (or 0.1)
6.6	10^{-2} (or 0.01)
10.0	10^{-3}
13.3	10^{-4}
16.6	10^{-5}
19.9	10^{-6}

$$I_n = I_0 2^{-n}$$

I_0 = dose rate or count rate at time zero

I_n = dose rate or count rate after n half-lives (or n half-value layers)

For estimating the value of I_n , for large values of n , there is a decrease by a factor of approximately 1,000 for each 10 half-lives.

* While these relationships hold exactly for half-lives, they may be used also for half-value layers with a fair degree of approximation, holding most closely for monoenergetic radiation and for narrow beams.

Section 2

GLOSSARY OF TERMS

Compiled by

HANSON BLATZ, *Director, New York City Office of Radiation Control, New York.*

Acknowledgments. Many of the terms listed are reproduced from the American Standard Glossary of Terms in Nuclear Science and Technology by special permission of the National Academy of Sciences. Others are from the "Radiological Health Handbook" of the U.S. Department of Health, Education and Welfare (Robert A. Taft Sanitary Engineering Center, Cincinnati, Ohio).

AEC Definitions. Definitions marked "[AEC]" are those appearing in the Atomic Energy Act of 1954, which makes their use mandatory in AEC negotiations, contracts, and communications. Although, in most cases, quite adequate prior definitions existed, these terms are creeping into the scientific language. The most common synonym is given parenthetically for some.

Radioactive Ores and Minerals. In Sec. 6, Tables 6-9, 6-10, and 6-11, a comprehensive glossary of these materials has been assembled by Dr. Truman Kohman. They have therefore been omitted from this section.

GLOSSARY OF TERMS

Compiled by Hanson Blatz

absorption

1. The process whereby the total number of particles or quanta emerging from a body of matter is reduced relative to the number entering, as a result of interaction of the particles with the body.

2. The process whereby the kinetic energy of a particle is reduced while traversing a body of matter. This loss of kinetic energy of corpuscular radiation is also referred to as moderation, slowing, or stopping.

3. The process whereby some or all of the energy of electromagnetic radiation is transferred to the substance on which it is incident or which it traverses.

4. For particulate radiation, energy is lost by collisions with electrons or nuclei. For photons, the reduction is due to the transfer of the energy to electrons by scattering and photoelectric processes and, at voltages greater than a million, by pair production.

Note: In the above sense this term is used interchangeably with attenuation. More specifically absorption refers to processes by which the radiation disappears or is transformed and not merely scattered.

absorption coefficient (μ)

A measure of the rate of decrease in intensity of a beam of photons or particles in its passage through a particular substance. The linear absorption coefficient μ_l is the fractional decrease in intensity per unit distance traversed, or $\mu_l = -dI/I dx$, where I is the intensity of the beam and x is the distance traversed. If μ_l is independent of x , then $I = I_0 e^{-\mu_l x}$, where I_0 is the initial intensity of the beam. The mass absorption coefficient μ_m is the fractional decrease in intensity per unit surface density. For a substance of density ρ , μ_m is equal to μ_l/ρ and hence is independent of the density.

absorption curve

A graph in which the intensity of the transmitted radiation is plotted as a function of thickness of the material traversed.

Usually the logarithm of the transmitted intensity is plotted vs. the surface density of absorber measured, for example, in g/cm². Most frequently the curve is determined for the purpose of characterizing the incident radiation as to its nature and energy. The form of the curve is somewhat dependent on the experimental arrangement.

accelerator

A device for imparting large kinetic energy to charged particles, such as electrons, protons, deuterons, and helium ions. Common types of accelerators are the cyclotron, betatron, linear accelerator, and Van de Graaff electrostatic accelerator. (See Particle Accelerators in Sec. 6.)

active

1. Fissionable (active material).
2. Radioactive (active sample).
3. The active part of a reactor is the core (active lattice).

active deposit

1. The radioactive decay products deposited on a surface exposed to radon, actinon, or thoron gas.
2. By extension, any radioactive decay products deposited on a surface exposed to a radioactive gas.
3. Any radioactive material deposited on a surface.

alpha emitter

A radionuclide that undergoes transformation by alpha-particle emission.

alpha particle (α)

1. A positively charged particle emitted from a nucleus and composed of two protons and two neutrons. It is identical in all measured properties with the nucleus of a helium atom.
2. By extension, the nucleus of a helium atom ($Z = 2$, $A = 4$), especially when it is in rapid motion as when artificially accelerated.

alpha ray (α ray)

A synonym for alpha particle.

anemia

A condition in which the blood is deficient in either quantity or quality. The deficiency in quality may consist of diminution of the amount of hemoglobin or of diminution of the number of red blood corpuscles.

angstrom (A)

A unit of length, used chiefly in expressing short wavelengths. It equals 10^{-10} meter or 10^{-8} centimeter.

annihilation

1. A process in which a pair of antiparticles meet and convert spontaneously into one or more photons; it is the inverse of pair production. The commonest example is the annihilation of an electron and a positron, the rest masses of which are converted into (usually) two 0.511-Mev photons according to the principle of mass-energy equivalence.
2. The conversion of rest mass into electromagnetic radiation.

annihilation radiation

Electromagnetic radiation produced by the union, and consequent annihilation, of a positron and an electron. Each such annihilation usually produces two, rarely

one or three, photons. These photons have properties identical with those of gamma rays, and accompany the decay of all positron-emitting radioactive substances. A positron and an electron are most likely to unite when their relative velocity is small; hence the energy available for the annihilation radiation will be that of the rest masses of the electron, $2m_0c^2$ ($= 1.02$ Mev), and the process will usually result in the production of two oppositely directed photons, each of energy 0.51 Mev.

area monitoring

Routine monitoring of the level of radiation or of radioactive contamination of any particular area, building, room, or equipment. Usage in some laboratories or operations distinguishes between routine monitoring and survey activities.

artificial radioactivity

The property of radioactivity induced in certain elements under controlled conditions.

atom

Smallest particle of an element which is capable of entering into a chemical reaction.

atomic energy

In popular usage, nuclear energy released in sufficient quantity to be of engineering interest.

atomic energy [AEC] *

All forms of energy released in the course of nuclear fission or nuclear transformation (nuclear energy).

atomic number (Z)

An integer that expresses the positive charge of the nucleus in multiples of the electronic charge e . It is the number of electrons outside the nucleus of a neutral atom and, according to present theory, the number of protons in the nucleus.

atomic weight (A)

A weighted mean of the masses of the neutral atoms of an element expressed in atomic-weight units.

attenuation

1. (Radiation theory) The reduction in the flux density, or power per unit area, with distance from the source; it may be due to absorption, to scattering, or to both processes.

2. (Nuclear physics) The reduction in the intensity of radiation upon passage through matter; in general, it is due to a combination of scattering and absorption.

The term is best used in a sense that excludes the geometric decrease of intensity with distance from the source (inverse-square effect); the attenuation then depends only on the nature of the radiation and of the material traversed.

attenuation factor

A measure of the opacity of a layer of material for radiation traversing it; the ratio of the incident intensity to the transmitted intensity. It is equal to I_0/I ,

* See page 2-1.

where I_0 and I are the intensities of the incident and emergent radiation, respectively. In the usual sense of exponential absorption ($I = I_0 e^{-\mu x}$), the attenuation factor is $e^{+\mu x}$, where x is the thickness of the material and μ is the absorption coefficient.

autoradiograph

(See *radioautograph*)

average life or mean life

The average of the individual lives of all the atoms of a particular radioactive substance. It is 1.443 times the radioactive half-life.

background

Ever-present effects in physical apparatus above which a phenomenon must manifest itself in order to be measured. "Background" can take various forms, depending on the nature of the measurement. In electrical measurements of radioactivity and nuclear phenomena, the term usually refers to those undesired counts or currents that arise from cosmic rays, local contaminating radioactivity, insulator leakage, amplifier noise, power-line fluctuations, and so on. In nuclear work and photographic emulsions, the term refers to developable grains unrelated to the tracks under investigation.

background radiation

Radiation arising from radioactive material other than the one directly under consideration. Background radiation due to the cosmic rays and natural radioactivity is always present. There may also be background radiation due to the presence of radioactive substances.

backscatter or backscattering

The deflection of particles or of radiation by scattering processes through angles greater than 90° with respect to the original direction of motion.

barn (b)

A unit of area used in expressing a nuclear cross section. 1 barn = 10^{-24} cm². Cross sections per atom are customarily measured in barns.

beam (see also *useful beam*)

A unidirectional or approximately unidirectional flow of electromagnetic radiation or of particles.

beta decay or beta disintegration

Radioactive transformation of a nuclide in which the atomic number changes by ± 1 and the mass number remains unchanged. Increase of atomic number occurs with negative beta-particle emission, decrease with positive beta-particle (positron) emission or upon electron capture.

beta emitter

A radionuclide that disintegrates by beta-particle emission.

beta particle (β ray)

A negative electron or a positive electron (positron) emitted from a nucleus during beta decay. The symbols β , β^- , and β^+ are reserved for electrons of nuclear origin.

beta ray

A synonym for beta particle.

betatron, induction accelerator

A device for accelerating electrons by means of magnetic induction.

Bev

The symbol for billion electron volt, or 10^9 ev.

biological effectiveness (of radiation)

(See *relative biological effectiveness*)

biological half-life

(See *half-life*)

biophysics

The study of phenomena of living organisms by physical methods. The study of physical phenomena exhibited by living organisms or parts thereof.

blood count (hematology)

Red count is the number of red corpuscles per cubic millimeter of blood. Total white count is the number of white corpuscles per cubic millimeter of blood. Differential white count is the number of each variety of white corpuscles in a count of 100. (The two most numerous varieties are polymorphonuclear leucocytes and lymphocytes.) Platelet count is the number of platelets per cubic millimeter of blood.

bone seeker

Any compound or ion which migrates in the body preferentially into bone.

boron chamber

An ionization chamber lined with boron or boron compounds and/or filled with a gaseous boron compound such as BF_3 .

boron counter tube

A counter tube filled with BF_3 and/or having electrodes coated with boron or boron compounds used for detecting slow neutrons by the (n, α) reaction of B^{10} .

bremsstrahlung

1. The production of electromagnetic radiation by the acceleration that a fast charged particle (usually an electron) undergoes when it is deflected by another charged particle (usually a nucleus). The spectral distribution is continuous, the well-known continuous X-radiation being a prominent example. For very energetic electrons (above about 50 Mev), the energy loss by radiation far exceeds that by ionization as a stopping mechanism in matter; this process is sometimes called "outer bremsstrahlung" in order to distinguish it from inner bremsstrahlung. A very weak electromagnetic radiation with a continuous spectral distribution is sometimes observed from beta-active substances; this is due to one or both types of bremsstrahlung.

2. Sometimes, the radiation resulting from the bremsstrahlung process.

build-up factor

The ratio of the intensity of X- or gamma radiation (both primary and scattered) at a point in an absorbing medium to the intensity of only the primary radiation. This factor has particular application for "broad-beam" attenuation. "Intensity" may refer to energy flux, dose, or energy absorption.

by-product material [AEC],* (radioactive isotope)

Any radioactive material (except special nuclear material) yielded in or made radioactive by exposure to the radiation incident to the process of producing or utilizing special nuclear material.

capture

A process in which an atomic or nuclear system acquires an additional particle; for example, the capture of electrons by positive ions, of electrons by nuclei, or of neutrons by nuclei. The gamma rays emitted after such capture are called "capture gamma rays."

carcinogenesis

Production of cancer.

carcinogenic

Capable of producing cancer.

collimator

A device for confining the elements of a beam within an assigned solid angle.

contamination, radioactive

1. A condition in which an undesirable radioactive substance is mixed with a desired substance.
2. A condition in which radioactive material has spread to places where it may harm persons, spoil experiments, or make products or equipment unsuitable or unsafe for some specific use.

count (radiation measurements)

The external indication of a device designed to enumerate ionizing events. It may refer to a single detected event or to the total registered in a given period of time. The term is loosely used to designate a disintegration, ionizing event, or voltage pulse.

count-rate meter

A device which gives a continuous indication of the average rate of ionizing events.

counter

A device for counting ionizing events. The term may refer to a complete instrument or loosely to the detector.

critical

In nuclear-science usage, this term is used to specify the mass, arrangement, or condition of a quantity of fissionable material such that it can sustain a chain

* See page 2-1.

reaction, e.g., a critical mass or a critical assembly. Prompt critical is capable of sustaining a chain reaction without the aid of delayed neutrons.

criticality

The condition of being critical. A reactor is described as having reached criticality when the chain reaction starts.

cross section, nuclear

The probability that a certain reaction between a nucleus and an incident particle or photon will occur. It is expressed as the effective "area" that the nucleus presents for the reaction.

curie (c)

1. The present definition of the curie is the unit of radioactivity defined as the quantity of any radioactive nuclide in which the number of disintegrations per second is 3.70×10^{10} .

2. An earlier definition of the curie was the quantity (grams) of radon in equilibrium with 1 gram of radium.

cyclotron

A device for accelerating charged particles to high energies by giving successive increments of energy from an alternating electric field between electrodes placed in a constant magnetic field.

daughter

A nuclide formed in the radioactive decay of another (called the parent). A synonym for decay product.

decay, radioactive

The spontaneous transformation with a measurable lifetime of a nuclide into one or more different nuclides. The process involves the emission from the nucleus of alpha particles, electrons, positrons, gamma rays, or the nuclear capture or ejection of orbital electrons, or fission. A synonym for radioactive disintegration.

decontamination

1. The removal of unwanted radioactive substances from a desired material, e.g., removal of fission products from plutonium or uranium.

2. The removal of undesired dispersed radioactive material from personnel, instruments, rooms, equipment, etc. In the case of physical objects this may involve thorough washing, often with chemical solutions; and in the case of fluids such as air, it may involve filtering and washing.

densitometer, photographic

Instrument for measuring photographic density.

density, photographic

The degree of darkening of photographic film; expressed as the logarithm of the opacity of exposed and processed film. Opacity is the reciprocal of transmission; transmission is the ratio of transmitted to incident intensity.

deuteron

The nucleus of a deuterium, or heavy hydrogen, atom ($Z = 1$, $A = 2$), the hydrogen isotope of mass number 2. It consists of a neutron and a proton.

discomposition

The process in which an atom is knocked out of its position in a crystal lattice by direct nuclear impact, e.g., by fast neutrons or by fast ions which have been previously knocked out of their lattice positions. The atom so displaced may eventually come to rest at an interstitial position or at a lattice edge.

discomposition effect

Change in chemical or physical properties resultant from discomposition. Also called "Wigner effect."

disintegration

A spontaneous nuclear transformation (radioactivity) characterized by the emission of energy from the nucleus. When numbers of nuclei are involved the process is characterized by a definite half-life.

1. Radioactive decay.

2. The transformation of one nuclide into one or more different nuclides by bombardment with high-energy particles such as alpha particles or helium ions, deuterons, protons, neutrons, or gamma rays.

dose or dosage

According to current usage, the radiation delivered to a specified area or volume or to the whole body. Units for dose specification are roentgens for X- or gamma rays, rems or equivalent roentgens for beta rays. In radiology the dose may be specified in air, on the skin, or at some depth beneath the surface; no statement of dose is complete without specification of location. In recent years there has been an increasing tendency to regard a dose of radiation as the amount of energy absorbed by tissue at the site of interest per unit mass. (See *rad*)

absorbed dose: The quantity of energy imparted to a mass of material exposed to radiation. (See *rad*)

air dose: X-ray or gamma-ray dose expressed in roentgens delivered at a point in free air. In radiologic practice it consists of the radiation of the useful beam and that scattered from surrounding air.

cumulative dose (radiation): The total dose resulting from repeated exposures to radiation of the same region, or of the whole body.

depth dose: The radiation dose delivered at a particular depth beneath the surface of the body. It is usually expressed as a percentage of surface dose or as a percentage of air dose.

exit dose: Dose of radiation at surface of body opposite to that on which the beam is incident.

integral dose or volume dose: A measure of the total energy absorbed by a patient or any object during exposure to radiation. According to British usage the integral dose of X- or gamma rays is expressed in gram-roentgens.

maximum permissible dose (MPD): Maximum dose of radiation which may be received by persons working with ionizing radiation.

median lethal dose (LD-50): Dose of radiation required to kill, within a specified period, 50 per cent of the individuals in a large group of animals or organisms.

percentage depth dose: Amount of radiation delivered at a specified depth in tissue, expressed as a percentage of the amount delivered at the skin.

permissible dose: The amount of radiation which may be received by an individual within a specified period with expectation of no significantly harmful result to himself.

skin dose: Dose at center of irradiation field on skin. It is the sum of the air dose and backscatter.

threshold dose: The minimum dose that will produce a detectable degree of any given effect.

tissue dose: Radiation dose received by a tissue in the region of interest. In the case of X-rays and gamma rays, tissue doses are expressed in roentgens. At the present time the rep is the generally accepted unit of tissue dose for other ionizing radiations. (*See dose; see also rad*)

tolerance dose: Synonym for "permissible dose." The latter is generally considered the preferable term.

efficiency (counters)

A measure of the probability that a count will be recorded when radiation is incident on a detector. Usage varies considerably; so that it is well to make sure which factors (window transmission, sensitive volume, energy dependence, etc.) are included in a given case.

electrometer

Electrostatic instrument for measuring the difference in potential between two points. Used to measure change of electric potential of charged electrodes resulting from ionization produced by radiation.

electron

Negatively charged particle which is a constituent of every neutral atom and also exists in the free state. Unit of negative electricity equal to 4.8×10^{-10} electrostatic unit or 1.6×10^{-19} coulomb. Its mass is 0.000549 atomic mass unit.

electron volt (ev)

A unit of energy equal to the energy gained by a particle having one electronic charge when it passes in a vacuum through a potential difference of 1 volt; $1 \text{ ev} = 1.60 \times 10^{-12} \text{ erg}$.

electroscope or electrometer

An instrument for detecting an electric charge by means of the mechanical forces exerted between electrically charged bodies. The distinction is usually made that, in an electroscope, the movement of a thin filament or foil is observed either directly or by its shadow on a scale, while in an electrometer the movement of the deflected body (generally by torsion) is indicated by a pointer on a scale. Also used for electronic devices to measure very small currents or voltages.

energy dependence

The characteristic response of a radiation detector to a given range of radiation energies or wavelengths, as compared with the response of a standard open-air chamber.

equilibrium, radioactive

Among the members of a radioactive series, the state which prevails when the ratios between the amounts of successive members of the series remain constant.

secular equilibrium : If a parent element has a very much longer half-life than the succeeding ones, so that there is no appreciable change in its amount in the time interval required for the later products to attain equilibrium, then, after the condition is reached, equal numbers of atoms of all members of the series disintegrate in unit time. This condition is never actually attained but is essentially established in such a case as radium and its series to radium D. The half-life of radium is about 1,600 years, of radon approximately 3.82 days, and of each of the subsequent members a few minutes. After about a month essentially the equilibrium amount of radon is present, and then for a long time all members of the series disintegrate the same number of atoms per unit time.

transient equilibrium : If the half-life of the parent is sufficiently short, so that the quantity present decreases appreciably during the period under consideration, but is still longer than that of successive members of the series, a stage of equilibrium will be reached after which all members of the series decrease in amount exponentially with the period of the parent. An example of this is radon, with a half-life of approximately 3.82 days, and the successive members of the series to radium D.

film badge

An appropriately packaged photographic film for detecting radiation exposure of personnel; usually dental-size X-ray film, worn on the person, and frequently combined with an identification badge. The badge may contain two or three films of different sensitivity, and it may contain a filter which shields part of the film from certain types of radiation.

film ring

A film badge in the form of a finger ring.

fission

The splitting of a nucleus into two more or less equal fragments. Fission may occur spontaneously or may be induced by capture of bombarding particles. In addition to the fission fragments, neutrons and gamma rays are usually produced during fission.

fission products (FP)

The nuclides produced by the fission of a heavy-element nuclide such as U-235 or Pu-239.

flow counter

(See *gas-flow counter*)

gamma ray (γ ray)

Short-wavelength electromagnetic radiation of nuclear origin with a range of wavelengths from about 10^{-8} to 10^{-11} cm, emitted from the nucleus.

gas amplification

The ratio of the charge collected in a counter tube or ionization chamber to the charge produced in the active volume by the preliminary ionizing event.

gas counter

A counter in which the sample is prepared in the form of a gas and introduced into the counter tube itself.

gas-flow counter

A counter in which an appropriate atmosphere is maintained in the counter tube by allowing a suitable gas to flow slowly through the volume.

gene

Fundamental unit of inheritance, which determines and controls hereditary transmissible characteristics. Genes are arranged linearly at definite loci on chromosomes.

geometry

The physical relationship and symmetry of the portions of an assembly. In neutron calculations, plane, cylindrical, and spherical geometries are arrangements having the corresponding symmetries. In cross-section measurements, good geometry refers to the case where very few detected neutrons have had only slight chance of suffering a collision in the target. In poor geometry almost all particles leaving the target are detected. Loosely, geometry factor, or geometry, refers to the experimental arrangement of source, counter, and other equipment.

grenz rays

X-rays produced in the region of 5 to 20 kv, but most usually from 8 to 12 kv.

guard ring

An auxiliary electrode in a counter tube or ionization chamber to control potential gradients, reduce insulator leakage, and/or define the sensitive volume.

half-life, biological

The time required for the body to eliminate one-half of an administered dose of any substance by regular processes of elimination. This time is approximately the same for both stable and radioactive isotopes of a particular element.

half-life, effective

Time required for a radioactive element fixed in the tissue of an animal body to be diminished 50 per cent as a result of the combined action of radioactive decay and biological elimination.

$$\text{Effective half-life} = \frac{\text{biological half-life} \times \text{radioactive half-life}}{\text{biological half-life} + \text{radioactive half-life}}$$

half-life, radioactive

Time required for a radioactive substance to lose 50 per cent of its activity by decay. Each radionuclide has a unique half-life.

half-thickness

The thickness of any given absorber that will reduce the intensity of a beam of radiation to one-half its initial value.

health physics

A term in common use for that branch of radiological science dealing with the protection of personnel from harmful effects of ionizing radiation.

hot

A colloquial term meaning highly radioactive.

hot laboratory

A laboratory designed for handling of radioactive materials where the strengths of radioactive sources are so high (say 50 millicuries and up) that special precautions in handling are necessary.

intensity

1. Of radiation, the energy of the number of photons, or of particles, passing through unit area perpendicular to the line of propagation at the point in question, in unit time. For parallel radiation, the area refers to a surface normal to the direction of propagation. In radiology, this term is often used incorrectly in the sense of dose rate.
2. Of any source of particles, the total number of particles emitted, or passing through, a unit area per unit of time.
3. Of radioactivity, a synonym for activity.

ion

A charged atom or molecularly bound group of atoms; also a free electron or other charged subatomic particle. An ion pair consists of a positive ion and a negative ion (usually an electron) having charges of the same magnitude and formed from a neutral atom or molecule by the action of radiation.

ion chamber

(See *ionization chamber*)

ionization

Any process by which a neutral atom or molecule loses or gains electrons, thereby acquiring a net charge; the process of producing ions.

ionization chamber

A device consisting of an enclosed or defined volume of gas between electrodes which may be charged or connected to a voltage source to produce an electric field throughout the volume, and intended for use in measuring quantity of ionizing radiation in terms of the electric charge associated with ions produced in the volume.

ionization chamber, air-wall (X- or gamma rays)

Ionization chamber in which the materials of the wall and electrodes are so selected as to produce ionization essentially equivalent to that in a free-air ionization chamber. This is possible only over moderate ranges of photon energies. Such a chamber is more appropriately termed an "air-equivalent ionization chamber." (See *ionization chamber, free-air*)

ionization chamber, free-air

An air-filled ionization chamber in which a delimited beam of radiation passes between the electrodes without striking them or other internal parts of the instrument, and so arranged as accurately to define a sensitive volume of such magnitude that the secondaries are produced only in air and are fully utilized. The electric field is maintained perpendicular to the electrodes in the collecting region; as a result the ionized volume can be accurately determined from the dimensions of the collecting electrode and the limiting diaphragm.

ionization chamber, integrating

An ionization chamber whose collected charge is stored in a capacitor for subsequent measurement.

ionization chamber, tissue-equivalent

An ionization chamber in which the material of the walls, electrode, and gas are so selected as to produce ionization essentially equivalent to that characteristic of the tissue under consideration. In some cases it is sufficient to have only tissue-equivalent walls, and the gas may be air, provided the air volume is small.

ionization counter

An ionization chamber which has no internal amplification by gas multiplication and which is used for counting ionizing particles.

isotope

One of several nuclides having the same number of protons in their nuclei, and hence belonging to the same element, but differing in the number of neutrons and therefore in mass number A , or in energy content (isomers).

isotope, radioactive

By common usage, any radioactive nuclide produced in a reactor (*see by-product material*) or in a particle accelerator. Correctly, it should include all the natural radioactive nuclides also.

kev

The symbol for 1 kilo-electron-volt, or 10^3 ev.

kilocurie (kc)

One thousand curies.

LD-50

(*See dose, median lethal*)

linear accelerator

A device for accelerating particles employing alternate electrodes and gaps arranged in a straight line, so proportioned that, when their potentials are varied in the proper amplitudes and frequency, particles passing through them receive successive increments of energy.

mean free path

The average distance l that a particle travels between successive collisions with the other particles of an ensemble.

mean life

(See *average life*)

median lethal dose (LD-50)

(See *dose, median lethal*)

megacurie (Mc)

One million curies.

Mev

The symbol for 1 million electron volts, or 10^6 ev.

microcurie (μc)

One-millionth of a curie.

millicurie (mc)

One-thousandth of a curie.

million electron volt (Mev)

A common unit of energy in nuclear science, equivalent to 10^6 ev.

milliroentgen (mr)

One-thousandth of a roentgen.

(Also *millirads and millirems*—see *rad, rep, and rem*)

moderation

The slowing down of a particle, usually a neutron, as a result of collisions with nuclei.

moderator

Material used in a nuclear reactor to moderate, i.e., slow down, neutrons from the high energies at which they are released.

monitor

An instrument to measure continuously or at intervals a condition that must be kept within prescribed limits, e.g., radioactivity in the coolant of a reactor or radiation levels in the working areas of the building. Also used as a verb.

monitoring

Periodic or continuous determination of the amount of ionizing radiation or radioactive contamination present in an occupied region, or in a person, as a safety measure for purposes of health protection.

area monitoring: Routine monitoring of the level of radiation or of radioactive contamination of any particular area, building, room, or equipment. Usage in some laboratories or operations distinguishes between routine monitoring and survey activities.

personnel monitoring: Monitoring any part of an individual, his breath, or excretions, or any part of his clothing.

n unit

That quantity of neutron radiation measured in a Victoreen condenser r meter that will produce the same amount of ionization as 1 roentgen of X-radiation.

neutron

Elementary nuclear particle with a mass approximately the same as that of a hydrogen atom and electrically neutral; its mass is 1.008986 mass units. Neutrons are commonly divided into subclassifications according to their energies as follows: thermal, around 0.025 ev; epithermal, 0.1 to 100 ev; slow, less than 100 ev; intermediate, 10^2 to 10^5 ev; fast, greater than 0.1 Mev.

nucleon

A constituent particle of the atomic nucleus; therefore, according to present theory, a proton or a neutron.

nucleus

The positively charged core of an atom, with which is associated practically the whole mass of the atom, but only a minute part of its volume.

nuclide

A species of atom characterized by the constitution of its nucleus.

pair production

The conversion of a photon into an electron and a positron when the photon traverses a strong electric field, such as that surrounding a nucleus or an electron.

parent

A radionuclide that upon disintegration yields a specified nuclide, the daughter, either directly or as a later member of a radioactive series.

person [AEC] *

1. Any individual, corporation, partnership, firm, association, trust, estate, public or private institution, group, government agency other than the AEC, and state or any political subdivision of, or any political entity within a state, any foreign government or nation or any political subdivision of any such government or nation, or other entity.

2. Any legal successor, representative, agent, or agency of the foregoing.

photon

A quantum of electromagnetic radiation. A photon of frequency ν has energy $h\nu$, where h is Planck's constant.

Planck's constant (h)

A universal constant h that has the value 6.624×10^{-27} erg sec. It is the proportionality factor that relates the energy E of a photon to its frequency ν , that is, $E = h\nu$.

pocket chamber

A small pocket-sized ionization chamber used for monitoring radiation exposure of personnel. Before use it is given a charge and the amount of discharge is a measure of the quantity of radiation received.

pocket dosimeter

A complete pocket-sized radiation instrument by means of which a quantity of radiation can be directly indicated.

* See page 2-1.

positron

A positive electron. Positrons are formed in the process of pair production, in the beta decay of many radionuclides, and in some more specialized processes.

production facility [AEC] *

1. Any equipment or device determined by rule of the AEC to be capable of the production of special nuclear material in such quantity as to be of significance to the common defense and security, or in such manner as to affect the health and safety of the public.
2. Any important component part especially designed for such equipment or device as determined by the AEC.

proton

A positively charged elementary particle of mass number 1 and charge equal in magnitude to the electronic charge e . It is one of the constituents of every nucleus; the number of protons in the nucleus of each atom of an element is given by the atomic number Z of the element.

quenching

The process of inhibiting continuous or multiple discharges in a counter tube which uses gas amplification.

r

Abbreviation for roentgen.

rad

One hundred ergs of absorbed energy per gram of absorbing material.

radiation

1. The emission and propagation of energy through space or through a material medium in the form of waves; for instance, the emission and propagation of electromagnetic waves, or of sound and elastic waves.
2. The energy propagated through space or through a material medium as waves; for example, energy in the form of electromagnetic waves or of elastic waves. The term radiation or radiant energy, when unqualified, usually refers to electromagnetic radiation; such radiation commonly is classified, according to frequency, as Hertzian, infrared, visible (light), ultraviolet, X-ray, and gamma ray. (See *photon*)
3. By extension, corpuscular emissions, such as alpha and beta radiation, or rays of mixed or unknown type, as cosmic radiation.

annihilation radiation: Photons produced when an electron and a positron unite and cease to exist. The annihilation of a positron-electron pair results in the production of two photons, each of 0.51 Mev energy.

background radiation: Radiation arising from radioactive material other than the one directly under consideration. Background radiation due to cosmic rays and natural radioactivity is always present. There may also be background radiation due to the presence of radioactive substances.

characteristic (discrete) radiation: Radiation originating from an atom following removal of an electron or excitation of the nucleus. The wavelength of the

* See page 2-1.

emitted radiation is specific depending only on the element concerned and the particular energy levels involved.

external radiation : Radiation from a source located outside the body.

internal radiation : Exposure to ionizing radiation when the radiation source is within the body as a result of deposition of radioelements in body tissues.

ionizing radiation : Any electromagnetic or particulate radiation capable of producing ions, directly or indirectly, in its passage through matter.

leakage radiation : The radiation which escapes through a protective shield.

primary radiation : All radiation coming directly from a source.

scattered radiation : Radiation which, during its passage through a substance, has been deviated in direction. It may also have been modified by an increase in wavelength. It is one form of secondary radiation.

secondary radiation : Radiation originating as the result of absorption of other radiation in matter. It may be either electromagnetic or particulate in nature.

stray radiation : Radiation not serving any useful purpose. It includes direct radiation and secondary radiation from irradiated objects.

radioactive equilibrium

(*See equilibrium*)

radioactive half-life

(*See half-life*)

radioactive series

A succession of nuclides, each of which transforms by radioactive disintegration into the next until a stable nuclide results. The first member is called the "parent," the intermediate members are called "daughters," and the final stable member is called the "end product."

radioactivity

Spontaneous nuclear disintegration with emission of corpuscular or electromagnetic radiations. The principal types of radioactivity are alpha disintegration, beta decay (negatron emission, positron emission, and electron capture), and isomeric transition.

radioautograph

Record of radiation from radioactive material in an object, made by placing its surface in close proximity to a photographic emulsion.

radiobiology

That branch of biology which deals with the effects of radiation on biological systems.

recombination

The return of an ionized atom or molecule to the neutral state.

recovery time

The time interval, after the initiation of a count, which must elapse before a counter is capable of delivering at the next ionizing event a pulse of substantially full size.

relative biological effectiveness (of radiation) (RBE)

The inverse ratio of the doses of two different radiations necessary to produce the same biological effect.

rem

(See *roentgen equivalent, man*)

rep

(See *roentgen equivalent, physical*)

r meter

Any ionization instrument calibrated to read in roentgens.

roentgen (r)

The quantity of X- or gamma radiation such that the associated corpuscular emission per 0.001293 gram of air produces, in air, ions carrying 1 esu of electricity of either sign.

roentgen equivalent, man, or mammal (rem)

The dose of any ionizing radiation that will produce the same biological effect as that produced by one roentgen of high-voltage X-radiation.

roentgen equivalent, physical (rep)

A unit of absorbed dose of radiation with a magnitude of 93 ergs per gram of absorbing material (usually soft tissue).

safety rod

A control rod capable of shutting down a reactor very quickly in case of failure of the ordinary control system.

saturation current

The ionization current which results when the applied potential is sufficient to collect all ions.

saturation voltage

The minimum value of applied potential required to produce saturation current.

scaler, scaling circuit

An electronic device which registers current pulses received over a given time interval.

binary scaler: A scaler whose scaling factor is 2 per stage.

decade scaler: A scaler whose scaling factor is a power of 10.

scattering

Change of direction of subatomic particle or photon as a result of a collision or interaction.

coherent scattering: Scattering of photons or particles in which there are definite phase relationships between the incoming and the scattered waves. Coherence manifests itself in the interference between the waves scattered by two or more scattering centers. Examples are the Bragg scattering of X-rays and of neutrons by the regularly spaced atoms in a crystal, for which constructive interference occurs only at definite angles, called Bragg angles.

compton scattering: The inelastic scattering of a photon through interaction with atomic electrons, accompanied by ejection of a recoil electron from the atom with which the interaction occurred. Compton-scattered photons carry away a fraction of the incident photon energy, ranging from an average of about 85 per cent of the initial energy for a 0.1-Mev photon to an average of about 30 per cent for a 10-Mev photon. Sometimes referred to as incoherent scattering.

elastic scattering: Scattering effected through the agency of elastic collisions and therefore with conservation of kinetic energy of the system. Rayleigh scattering is a form of elastic scattering.

incoherent scattering: Scattering of photons or particles in which the scattering elements act independently of one another so that there are no definite phase relationships among the different parts of the scattered beam. The intensity of the scattered radiation at any point is obtained by adding the intensities of the scattered radiation reaching this point from the independent scattering elements.

inelastic scattering: Scattering effected by inelastic collisions, such as are involved in the Compton effect.

modified scattering: Used synonymously with Compton scattering.

multiple scattering: Scattering of a particle or a photon in which the final displacement is the vector sum of many, usually small, displacements.

plural scattering: Scattering of a particle or a photon in which the final deflection is the vector sum of a small number of displacements.

Rayleigh scattering: The elastic scattering of a photon without loss of photonic energy. Sometimes referred to as coherent scattering.

single scattering: The deflection of a particle from its original path owing to one encounter with a single scattering center in the material traversed.

unmodified scattering: Used synonymously with Rayleigh scattering.

scintillation

A flash of light produced in a phosphor by an ionizing event.

scintillation counter

The combination of phosphor, photomultiplier tube, and associated circuits for counting scintillations.

self-absorption

The absorption of radiation by the body of the substance emitting the radiation.

source material [AEC] *

1. Uranium, thorium, or any other material which is determined by the AEC to be source material.

2. Ores containing one or more of the foregoing materials, in such concentration as the AEC may by regulation determine from time to time.

special nuclear material [AEC] *

1. Plutonium, uranium enriched in the isotope 233 or in the isotope 235, and any other material which the AEC determines to be special nuclear material, but does not include source material.

* See page 2-1.

2. Any material artificially enriched by any of the foregoing, but does not include source material (fissionable material).

specific activity

1. The activity of a radioisotope of an element per unit weight of element present in the sample.

2. The activity per unit mass of a pure radionuclide.

3. The activity per unit weight of any sample of radioactive material.

Note: Specific activity is commonly given in a wide variety of units (e.g., millicuries per gram, disintegrations per second per milligram, counts per minute per milligram, etc.).

specific ionization

The number of ion pairs formed per unit distance along the track of an ion passing through matter.

stopping power

A measure of the effect of a substance upon the kinetic energy of a charged particle passing through it. The relative stopping power is the ratio of the stopping power of a given substance to that of a standard substance, commonly aluminum, oxygen, or air.

survey instrument

A portable instrument, used for detecting and measuring radiation under varied physical conditions. The term covers a wide range of devices utilizing most of the detection methods defined elsewhere.

synchrocyclotron or f-m cyclotron

A cyclotron in which the radiofrequency of the electric field is frequency-modulated to permit the acceleration of particles to relativistic energies.

synchrotron

A device for accelerating particles, ordinarily electrons, in a circular orbit in an increasing magnetic field by means of an alternating electric field applied in synchronism with the orbital motion.

tissue-equivalent material

Material made up of the same elements in the same proportions as they occur in some particular biological tissue. Such material is important in the construction of ionization chambers for neutron measurement. In some cases the equivalence may be brought about sufficiently closely without exact duplication of the elemental composition of the tissue. (*See atomic number*)

useful beam

That part of the primary radiation which passes through the aperture, cone, or other collimator.

utilization facility [AEC] * (reactor or pile)

1. Any equipment or device except an atomic weapon, determined by rule of the AEC to be capable of making use of special nuclear material in such quantity as to

* See page 2-1.

be of significance to the common defense and security, or in such manner as to affect the health and safety of the public, or peculiarly adapted for making use of atomic energy in such quantity as to be of significance to the common defense and security, or in such manner as to affect the health and safety of the public.

2. Any important component part especially designed for such equipment or device as determined by the AEC.

Wigner effect

The discomposition effect.

X-rays

Electromagnetic radiation of wavelengths less than about 100 Å, produced: (1) when electrons striking a target lose kinetic energy in passing through the strong electric fields surrounding the target nuclei, thus giving rise to bremsstrahlung and resulting in a continuous X-ray spectrum; (2) by the transitions of atoms from *K*, *L*, ... energy states to lower-energy states, thus giving rise to characteristic X-rays.

Section 3

EXPOSURE STANDARDS AND RADIATION PROTECTION REGULATIONS

By

LAURISTON S. TAYLOR, *Chairman of the National Committee on Radiation Protection; Chief of the Atomic and Radiation Physics Division, National Bureau of Standards, Washington, D.C.*

SAUL J. HARRIS, *Assistant Manager of Technical Services, Atomic Industrial Forum, Inc., New York, N.Y.*

National Bureau of Standards Handbooks.....	3-2
Exposure Standards and Protection Regulations.....	3-2
Early History of Protection Standards.....	3-3
Basic Handbooks.....	3-11
Government and Nongovernment Agencies Establishing Radiation Standards and Regulations.....	3-31
Federal Regulations Applying to the Transportation of Radioactive Materials.....	3-42

EXPOSURE STANDARDS AND RADIATION PROTECTION REGULATIONS

Lauriston S. Taylor and Saul J. Harris

NATIONAL BUREAU OF STANDARDS HANDBOOKS

(Obtainable from Superintendent of Documents,
Government Printing Office, Washington 25, D.C.)

- H42 Safe Handling of Radioactive Isotopes, 1949
- H48 Control and Removal of Radioactive Contamination in Laboratories, 1951
- H49 Recommendations for Waste Disposal of Phosphorus-32 and Iodine-131 for Medical Users, 1951
- H51 Radiological Monitoring Methods and Instruments, 1952
- H52 Maximum Permissible Amounts of Radioisotopes in the Human Body and Maximum Permissible Concentrations in Air and Water, 1953
- H53 Recommendations for the Disposal of Carbon-14 Wastes, 1953
- H54 Protection against Radiations from Radium, Cobalt-60, and Cesium-137, 1954
- H55 Protection against Betatron-Synchrotron Radiations up to 100 Million Electron Volts, 1954
- H56 Safe Handling of Cadavers Containing Radioactive Isotopes, 1953
- H58 Radioactive-waste Disposal in the Ocean, 1954
- H59 Permissible Dose from External Sources of Ionizing Radiation, 1954
- H60 X-ray Protection (Revision of H41)
- H61 Regulation of Radiation Exposure by Legislative Means, 1955
- H62 Report of the International Commission on Radiological Units and Measurements, 1957
- H63 Protection against Neutron Radiation up to 30 Million Electron Volts, 1957
- H65 Safe Handling of Bodies Containing Radioactive Isotopes (Revision of H56)
- H66 Safe Design and Use of Industrial Beta-ray Sources, 1958 (Recommendations of ASA Z54 Sectional Committee)

EXPOSURE STANDARDS AND PROTECTION REGULATIONS

Normally, one thinks of a "standard" as something firm, well understood, inflexible, accurately known, and reproducible.¹ **Radiation protection "standards"** involve many unknowns and many uncertainties. It should be understood that

standards as presented by many authorities must be considered essentially goals or objectives. A radiation protection standard is established primarily to determine a numerical value for the limits of radiation dose which the individual or whole population can receive without encountering biological risks incommensurate with the benefits to be expected from the use of the radiation source. These numerical values are more in the nature of a means of achieving a goal than the goal itself.

It is necessary that a standard be established by means of which an environmental situation can be evaluated and which will permit, with some degree of certainty, assurance that any individual in that environment will not experience injury to himself—or his progeny. As such, the concept of a **maximum permissible dose** has evolved as a basic biological standard for any individual. Almost all other standards for controlling the exposure of human beings to radiation are based on maintaining the received dose within these permissible limits. It is hoped that greater study will eliminate various uncertainties and unknowns concerning the basic standard, at which time the standard dose limit will be established as a fundamental number in biophysics.

EARLY HISTORY OF PROTECTION STANDARDS

Soon after its discovery, radiation was recognized as a hazard. Organized efforts to understand and curtail these hazards were begun in the 1920s (Taylor, Lauriston S., *The Basis for Standards for Radiation Protection*, *J. Wash. Acad. Sci.*, vol. 46, pp. 69-77, 1956). The first standard adopted, in 1928, was a uniform acceptable unit of dose—the roentgen (defined in Sec. 2). Until then, protection efforts were of necessity purely qualitative. Radiation treatments were previously expressed in terms of fractions of an **“erythema” dose**—the amount of radiation which would cause a “well-defined” reddening of the skin. This was a very uncertain factor, dependent upon the radiation energy, dose rate, size of irradiated field, and scattering—and of course, individual variations in radiation sensitivity.

In 1925, Mutscheller (*Am. J. Roentgenol. Radium Therapy*, vol. 13, pp. 65-70, 1925) suggested the standard of 1/100 of an erythema dose in 30 days. In 1934, the limit of 0.2 r/day was chosen by the International Commission on Radiological Protection (*Recommendations for X-ray and Radium Protection*, *Radiology*, vol. 23, pp. 682-685, 1934). In 1936, 0.1 r/day was adopted in the United States, to provide an additional safety factor (“X-ray Protection,” NBS Handbook 20, GPO, 1936). The later figure remained the basic standard for radiation protection through the Manhattan project days—until 1948. In 1948, a basic figure of 0.3 r/week (“Permissible Dose from External Sources of Ionizing Radiation,” NBS Handbook 59, GPO, 1954), primarily for gamma rays and moderate-energy X-rays, was recommended by the National Committee on Radiation Protection and adopted internationally in 1950 (*Recommendations of the International Commission on Radiological Protection*, *Brit. J. Radiol., Suppl.* 6, 1954). All other values or standards for radiation limits until 1956 related to this basic weekly dose limit.

In 1956, the International Commission on Radiological Protection and the National Committee on Radiation Protection and Measurements recommended a further lowering of the maximum permissible dose (*Radiology*, vol. 70, p. 261, February, 1958). The new levels amount to a reduction by a factor of 3 in the

3-4 EXPOSURE STANDARDS AND PROTECTION REGULATIONS

values before used. They also recommend a further reduction by a factor of 10 for the nonoccupational exposure of people outside controlled areas. These changes will be discussed in more detail below.

The **International Commission on Radiological Protection (ICRP)** is one of two permanent commissions operating under the auspices of the International Congress of Radiology. The Congress was organized in 1925 to serve radiological and medical groups from all countries of the world, with delegates from radiological societies and national standardization laboratories from each country. The Protection Commission was first organized in 1928 and has held meetings during each succeeding Congress. Its first protection recommendations were published in 1928.

Committees at present operating under the International Commission on Radiological Protection include:

1. Permissible Dose for External Radiation
2. Permissible Dose from Internal Radiation
3. Protection against X-rays Generated at Potentials up to 3 Million Volts
4. Protection against X-rays above 3 Million Volts, Beta-rays, Gamma-rays, and Heavy Particles—Including Neutrons and Protons
5. The Handling and Disposal of Radioactive Isotopes

In 1956, the **World Health Organization (WHO)** formally recognized the ICRP as a Nongovernmental Participating Organization, to provide technical guidance and recommendations in the field of radiation hazards. The WHO will act as agent in implementing these throughout the world.

The **International Commission on Radiological Units and Measurements (ICRU)** is the other of the two permanent Commissions under the International Congress of Radiology. Established in 1925, the ICRU has attempted to define basic units of radiation dose utilizing basic physical units, rather than trying to build around techniques used for measurement. The Commission, in 1953 (*Recommendations of the International Commission on Radiological Units, Am. J. Roentgenol. Radium Therapy Nuclear Med.*, vol. 71, pp. 139-142, 1954) and in 1956 ("Report of the International Commission on Radiological Units and Measurements," NBS Handbook 62, GPO, 1957) considered that the roentgen, in view of its long-established usefulness, should continue to be recognized as the unit of X- and gamma-ray exposure dose, but at the same time recommended a unit of absorbed dose called the rad.

The **National Committee on Radiation Protection and Measurements (NCRP)** was originally known as the Advisory Committee on X-ray and Radium Protection. It was formed in 1929 upon the recommendation of the International Commission on Radiological Protection and is managed by the National Bureau of Standards, U.S. Department of Commerce. Sponsoring organizations represented on the parent committee include:

- American College of Radiology
- American Dental Association
- American Industrial Hygiene Association
- American Medical Association
- American Radium Society

American Roentgen Ray Society
American Veterinary Medical Association
Health Physics Society
International Association of Governmental Labor Officials
National Bureau of Standards
National Electrical Manufacturers Association
Radiological Society of North America
U.S. Air Force
U.S. Army
U.S. Atomic Energy Commission
U.S. Navy
U.S. Public Health Service

In the preparation of its recommendations beginning in 1946, the National Committee on Radiation Protection and Measurements gave long and serious consideration to a permissible-dose limitation making allowance for **genetic damage**. Such a standard, however, was not specifically included in its recommendations (NBS Handbook 59), since at the time there was very little quantitative information on human genetics. Consideration was also given to the effect of radiation exposure on the average **life expectancy** but here also the information was even more meager and specific allowance was not made for this effect.

By 1956, ten years later, a substantial increase in our knowledge of both these effects had been developed, and quantitative recognition of the problem was demonstrated in the 1956 recommendations of both the ICRP (*Radiology*, vol. 70, p. 261, February, 1958) and the NCRP (*Radiology*, vol. 68, p. 260, 1957). For genetic reasons, an age limitation on radiation exposure was introduced. The new limitations stipulate that the total occupational dose up to the age of thirty should not exceed approximately 50 rems, and up to the age of forty an additional 50 rems. Beyond age forty, the genetic considerations are relatively unimportant, but because of the possible effect of radiation on life span, it was considered wise to continue the same rate of exposure, namely, approximately 5.0 rems per year. Thus, a person receiving his maximum permissible occupational exposure during his working lifetime, up to age seventy, would accumulate of the order of 250 rems as compared with a total accumulation of about three times this amount under the levels permitted heretofore.

The ICRP and NCRP have in the past given serious thought to recommending a **national registry of radiation dose** received by each individual during occupational and medical exposure (*Physics in Medicine and Biology*, vol. 2, p. 107, 1957). For reasons of practicality, no group has pushed this point of view. No limits have been set for radiation doses receivable by any individual during the course of radio-diagnosis or radiotherapy.

In 1956, the scope of the NCRP was again broadened to include the development of recommendations relative to radiation standards and measurements in the medical radiological field. The present subcommittees are as follows:

1. Permissible Dose for External Sources
2. Permissible Internal Dose
3. X-rays up to 2 Million Volts

3-6 EXPOSURE STANDARDS AND PROTECTION REGULATIONS

4. Heavy Particles (Neutrons, Protons, etc.)
 5. Electrons, Gamma Rays, and X-rays above 2 Million Volts
 6. Handling of Radioactive Isotopes in Fission Products
 7. Monitoring Methods and Instruments
 8. Waste Disposal and Decontamination
 9. Protection against Radiations from Radium, Cobalt-60, and Cesium-137
- Encapsulated Sources
10. Regulation of Radiation Exposure
 11. Incineration of Radioactive Wastes
 12. Radiation Exposure under Emergency Conditions
 13. Safe Handling of Radioactive Cadavers
 14. Protection against High-energy and High-intensity Electrons
 - M1. Standards and Measurement of Radioactivity for Radiological Use
 - M2. Standards and Measurement of Radiological Exposure Dose
 - M3. Standards and Measurement of Absorbed Radiation Dose
 - M4. Relative Biological Effectiveness

The recommendations made by subcommittees of the NCRP have been published in **National Bureau of Standards (NBS) Handbooks** published by the U.S. Department of Commerce and made available through the Superintendent of Documents, Government Printing Office, Washington, D.C. (see list on page 3-2).

NBS Handbook 41—**Medical X-ray Protection up to Two Million Volts** (1949)

This handbook, issued in March, 1949, superseded NBS Handbooks 15 and 20 and is now, in turn, superseded by NBS Handbook 60, "X-ray Protection" (see below).

NBS Handbook 42—**Safe Handling of Radioactive Isotopes** (1949)

This handbook, issued September, 1949, was under revision at press time.

NBS Handbook 48—**Control and Removal of Radioactive Contamination in Laboratories** (1951)

Although published several years ago, the recommendations of this handbook still remain valid in most respects. Some of the material contained in it will also be included in the revised version of NBS Handbook 42.

A section on contamination of the skin gives permissible levels of contamination, although it must be recognized that these are subject to change as our knowledge improves. It also includes some general descriptions regarding decontamination procedures and the treatment of wounds which may be contaminated. Information is given with regard to levels of contamination and decontamination procedures for clothing and bedding, and for laboratory tools and glasswares. In the latter section there is some discussion regarding measures that should be taken to prevent undue contamination. Finally, there is a chapter on the handling of emergency situations.

This develops a general background on how to avoid emergencies by such means as the containment of spills and the avoidance of dust, mists, fumes, gases, etc. It also includes procedures to be taken in the case of fire. The general procedures to be followed in case of accidents are outlined. The appendix containing information on decontaminating detergents and wetting agents should probably be

regarded as obsolescent since there have been rapid advances in this field since the publication of the handbook.

NBS Handbook 49—Recommendations of Waste Disposal of Phosphorus-32 and Iodine-131 for Medical Users (1951)

Implicit in its title, this handbook applies mainly to users of relatively small quantities of radioactive material. By and large, most radioactive phosphorus or iodine, particularly as used in hospitals, may be disposed of in the sanitary sewer. Directions are given for the amounts of material that may be disposed of during a single 24-hr period (Table 3-1). Where the amounts are too large for simple disposal,

Table 3-1. Permissible Activities in Millicuries for Single Disposal Events

Number of beds	Number of people *	Toilet disposal, mc		Batch bottle disposal day, mc †
		Day	Night	
25	50	1-4	1-3	1
50	100	2-4	1-4	2
100	200	2-5	2-4	4
200	400	2-6	2-5	8
300	600	2-8	2-6	12
500	1,000	4-11	2-8	20
1,000	2,000	6-20	4-13	40

* It is assumed that the hospital population is equal to twice the number of beds.

† Batch discharge of a full 1-gal bottle (add tap water if necessary) emptied into a sink, not into a toilet.

directions are given with regard to storage or slow uniform disposal. It is shown how to calculate the permissible amounts for disposal dependent upon the type and size of sewerage system. Calculations of sewerage contamination in the disposal of radioactive iodine and phosphorus are given in the appendix.

NBS Handbook 51—Radiological Monitoring Methods and Instruments (1952)

With the rapidly advancing art, any treatment of this subject becomes obsolescent fairly rapidly. This handbook has tried to avoid discussion of specific instruments and deals mainly with broad principles. For that reason, it can still be regarded as of general value except in certain minor areas. A general discussion is given of radiation detectors and measuring instruments for beta-gamma radiations, alpha rays, and neutrons. Procedures and problems rather than specific techniques are the main topics of discussion. Since publication of the handbook, there have been important advances in the use of scintillation counters and in the measurement of neutrons, in neither of which subjects is the handbook considered to be up to date as far as techniques are concerned.

A section on personnel-monitoring instruments covers the general instrumentation requirements for the measurement or detection of all types of radiation. Included in this is some discussion regarding the properties of radiation and their influence

upon the measurements. This is followed by specific instrument requirements which should be embodied in instrument specifications.

In the section on Radiation Survey Methods, the general details and procedures of surveys are outlined. This section also included some discussion of particular problems that may exist in certain types of installations. Much of this material is drawn from other handbooks but is included here as illustrative of the type of situation that should be looked for during the survey of a radiation installation.

The appendix, in addition to giving various useful data relative to the measurement of radiation, also discusses techniques and equipment for air sampling, analysis of water, and the counting of specimens for assay purposes.

NBS Handbook 52—Maximum Permissible Amounts of Radioisotopes in the Human Body and Maximum Permissible Concentrations in Air and Water (1953)

As in the case of NBS Handbook 59, most of the material contained in this handbook was developed several years prior to its publication. Recommendations in this handbook go back basically to the fundamental dose-limit figure of 0.3 rem per week, which in turn is believed to be closely related to the maximum permissible amount of Radium-226 in the body. Extensive information based on the radium-poisoning cases in New Jersey, many years ago, provides a substantial body of information against which the effect of other radioactive isotopes in the body may be compared. Of course the exact comparison with radium can be made only with those elements which behave similarly in the body. Quantitative information in these recommendations comes from a large number of sources, including experiments with animals and experience, in a limited number of cases, with man. Since much of the material has had to be calculated rather than be based on specific experimental evidence, it has been necessary to agree on the characteristics of the standard man.

All the calculations on this report are based on a continuous exposure of 24 hr a day, 7 days a week, and 49 weeks per year. Therefore, if the recommendations are applied to occupational exposures, a corresponding allowance should be made.

In arriving at the final recommendation for permissible amounts of radioactive isotopes in the body and concentration in air and water, a large number of factors had to be considered. These were as follows: (1) quantities of radioactive materials available, (2) initial body retention, (3) fraction of radioactive material going from the blood to the critical body tissues, (4) radiosensitivity of the tissues, (5) size of critical organs, (6) essentiality of the critical organs to the proper function of the body, (7) biological half-life, (8) radioactive half-lives of intermediate length, (9) energy of the radiation produced by the radioisotope, and (10) the specific ionization and attenuation of energy in tissue. Needless to say, some of these factors are not known with any high degree of accuracy for many of the radioactive isotopes. As a result of this, the tables of concentrations must remain under continual review, and it may be expected that they will change from time to time. They have already been changed several times.

The handbook also gives the essential information and methods of calculation to show how the final numbers were reached and how calculations may be made for additional isotopes as new information becomes available.

Tables for the maximum permissible amounts of radioactive isotopes in the total body and the maximum permissible concentration in air and water are given in Table 3-3. Since the tables of factors used for calculating these values are too extensive, it is recommended that reference be made to the handbook and to Sec. 14.

NBS Handbook 53—Recommendations for the Disposal of Carbon-14 Wastes (1953)

This handbook follows somewhat the pattern of the corresponding handbook for radioactive phosphorus and radioactive iodine. However, the problem of disposal of radioactive wastes containing Carbon-14 is somewhat simpler than for many other materials. Specific limitations with regard to disposal by sewer, isotopic dilution, incineration, atmospheric dilution, garbage, and burial are described. The handbook includes the necessary information to calculate the safe disposable amounts.

NBS Handbook 54—Protection against Radiations from Radium, Cobalt-60, and Cesium-137 (1954)

This handbook supersedes NBS Handbook 23 (1938) on Radium Protection. The scope has been considerably broadened to include other commonly used isotopes in sealed sources. The handbook is written essentially in code form since much of the material contained in it is well established as a result of long experience.

One section includes recommendations regarding the equipment and facilities for the handling, storage, and transportation of the radioactive material, including detailed discussions on such items as forceps, clamping devices, and magnetic or pneumatic handling devices. A subsection on transportation facilities gives recommendations with regard to movement by private automobile or public transport.

The section on medical applications deals primarily with clinical problems and the application of the sources to the patients. A second section on medical applications relates essentially to teletherapy problems where large sources are used externally to the patient. Requirements for the design of teletherapy apparatus are given, including the necessary safety devices. In a section on nonmedical applications, recommendations are given with regard to calibration sources, the application of protection measures in research institutions, and some discussion with regard to the use of mixed sources for civil-defense purposes. A section on protection surveys and personnel monitoring outlines the details that should be looked for in surveys, together with information and warning signs, the detection of leakage, personnel monitoring, etc.

Another section gives recommendations regarding proper working conditions. This includes the designation and responsibilities of the radiological safety officer and the matter of physical examinations for employees. A section on accidents entailing radiation hazards deals with problems of spillage and the care of contaminated persons. This covers special problems that may be involved in the case of a radium spill. It also covers matters of decontamination and the disposal of radioactive wastes that may be involved in an accident. An appendix covers the details of the encapsulation and sealing of sources. Special details with regard to radon and radon and radium are covered. A second appendix lists the shipping rules of the ICC, Post Office Department, and Civil Aeronautics Board, and a final

3-10 EXPOSURE STANDARDS AND PROTECTION REGULATIONS

appendix gives the necessary barrier-design data together with the computations necessary for selecting the proper barrier.

NBS Handbook 55—Protection against Betatron-Synchrotron Radiations up to 100 Million Electron Volts (1954)

The problems of protection in the energy region above a few million electron volts present considerable difficulties, because of both the lack of experimental knowledge and the difficulties involved in the measurement of the radiation, particularly for survey purposes. This handbook outlines the philosophy of radiation measurement and provides detail as to the properties of these very high energy X-rays. Specific recommendations give information for protection against operating hazards such as electrical protection, warning and safety devices, and fire protection.

Protection against radiation hazards includes a discussion of the units and methods of measurement. These methods apply to X-rays, electrons, and the neutrons that may be produced secondarily. Since all these radiations may be involved in a betatron or synchrotron operation, special techniques must be applied for their proper measurement. A section on surveys outlines as much as is presently known about these problems. The handbook as a whole includes a considerable amount of basic technical information relative to the whole problem of high-energy-radiation measurement.

NBS Handbook 56—Safe Handling of Cadavers Containing Radioactive Isotopes (1953)

Superseded—see Handbook 65.

NBS Handbook 58—Radioactive-waste Disposal in the Ocean (1954)

This handbook discusses the broad question of the disposal of radioactive wastes at sea. Since there is very little quantitative information about the problems involved, much of the report deals with general discussion of the nature of the ocean bottom and what is likely to happen to radioactive materials deposited there. It is recognized that, since this may involve international considerations, it is unlikely that large-scale disposal of radioactive wastes in the ocean will be undertaken in the near future. A summary of the recommendations for disposal in the ocean is given; this includes the design and necessary characteristics of the waste containers.

NBS Handbook 60—X-ray Protection (1955)

This is the latest version of the handbook on X-ray protection relative primarily to problems in the medical field. It is for the most part equally applicable to problems in the industrial field. It is an outgrowth of the earlier handbooks on the same subjects, Nos. 15 (1934), 20 (1936), and 41 (1949).

This handbook is written essentially in the form of a code and may be used as such with little or no modification. It covers in detail problems of planning of new installations and surveying and inspection of new or existing installations. It also goes into considerable detail with regard to the working conditions in medical in-

installations. A section includes a considerable amount of structural detail necessary for the design of protective barriers. Special rules are given for medical fluoroscopic installations, X-ray installations, medical radiographic X-ray installations, dental radiographic installations, mobile diagnostic equipment, fluorographic equipment, and therapeutic installations operating up to about 3 Mev. The section on electrical protection in the earlier handbook has been eliminated since this is now considered a relatively unimportant problem with the widespread use of shockproof X-ray equipment.

In this handbook, the design of radiation protection takes into consideration tube potential, work load, occupancy factors, and the number of milliamperere minutes of operation per week for the specific equipment under consideration. Full use of these factors permits very tight design and the minimization of radiation shielding and barriers. In view of the recent limitation placed on doses up to age thirty, an additional factor of 3 should be added into the calculations if it is expected that exposed individuals will be employed continuously in such installations.

NBS Handbook 63—Protection against Neutron Radiation up to 30 Million Electron Volts (1957)

This handbook discusses the broad question of protection against neutrons over a wide energy range. It includes a digest of the current status of physical and biological information and the units used for measuring neutrons. There is also discussion of the interaction of neutrons and tissue and the relative biological effectiveness. There is a section on protection installation and the operation of neutron sources, discussing the various shielding and mechanical problems of providing protection. A series of suggested rules for protection against neutron radiation are provided.

NBS Handbook 65—Safe Handling of Bodies Containing Radioactive Isotopes (1958)

This is an extension and updating of NBS Handbook 56.

This handbook is designed primarily for use in hospitals or mortuaries where it may be necessary to handle cadavers containing radioactive material. The initial sections outline the general problems that may be involved and are followed by an outline of conditions where either an autopsy or no autopsy may be permitted. The handbook also includes some special recommendations regarding the treatment of an accident or injury to an operator during an autopsy and the handling of contaminated clothing or instruments. Special precautions to be taken during the course of cremation are also discussed. In the appendix, there is given an acceptable form for a radioactivity report that may accompany a cadaver.

BASIC HANDBOOKS

The two handbooks having the most basic influence on our whole problem of radiation protection are those giving the **permissible dose from internal and external sources of ionizing radiation**. All the basic data in the other handbooks are in conformity with the permissible-dose requirements as specified in these two reports (NBS Handbooks 52 and 59). While the two reports were actually published

3-12 EXPOSURE STANDARDS AND PROTECTION REGULATIONS

in 1953 and 1954, the basic philosophy had been fairly clearly developed as early as 1948, and interim reports were available to the other subcommittees during the course of preparation of their respective recommendations.

NBS Handbook 59—Permissible Dose from External Sources of Ionizing Radiation (1954)

This handbook initially introduced a number of basic and far-reaching decisions as follows: (1) to lower the earlier existing permissible dose from 0.1 r/day by a factor of about 2 and to express it in terms of a weekly limit; (2) to take the blood-forming organs as the most critical tissue and to apply the permissible limit of 0.3 r/week to these organs; (3) to consider the skin as a critical organ and to set the permissible dose for it at 0.6 r/week at the depth of 7 mg/cm²; (4) to allow a permissible dose for persons over forty-five years of age double that for younger adults (now superseded); (5) to allow larger weekly doses for the hands and feet (1.5 r/week); (6) to make suitable recommendations about accidental exposure involving a single dose that might be as large as 25 r; (7) to recommend relative biological effectiveness (RBE) values of 1 for X-rays, gamma rays, and beta rays, 5 for thermal neutrons, 10 for fast neutrons, and 10 for alpha particles.

While genetic effects of radiation were by no means overlooked in the preparation of the recommendations contained in this report, prime consideration was given to the effect of radiation on the individual himself rather than on his progeny. Basically, the **maximum permissible dose** as defined in this handbook is "that dose of radiation which in the light of present knowledge is not expected to cause appreciable bodily harm to a person at any time during his life-time." Included in this is, of course, the effect on the life span of the individual. At the time the initial report was written, there was little quantitative knowledge on this subject, and, in fact, there is not a great deal more of a direct nature at present. That there is some effect is not questioned, but there is so little quantitative information on the subject that it was not felt justifiable to attempt to quantify this in terms of permissible dose. The report contains considerable discussion regarding specific radiation effects on certain organs, such as the eye, the gonads, and skin. Wherever quantitative information was available, this was taken into consideration in setting special limits for these organs.

The Apr. 15, 1958, recommendations for the MPD to external and internal radiation are given below:

MAXIMUM PERMISSIBLE RADIATION EXPOSURES TO MAN

INTRODUCTION

On January 8, 1957, the National Committee on Radiation Protection and Measurements issued a Preliminary Statement setting forth its revised philosophy on Maximum Permissible Radiation Exposures to Man. Since that time several of the NCRP subcommittees have been actively studying the necessary revisions of their respective handbooks. These studies have shown the need for (1) clarification of the earlier statement and (2) modification or extension of some of the concepts in that statement. Furthermore, the International Commission on Radiological Protection has made minor changes in their recommendations. Accordingly the NCRP has prepared a set of guides,

given below, that will assure uniformity in the basic philosophy to be embodied in the various handbooks. Since many of the handbooks are followed closely in planning radiation operations in the United States, and since the modification of a handbook may require many months of effort, it seems wise to make the over-all guiding principles available in advance of the reissuance of the revised handbooks. These guides are not designed to take the place of any of the handbooks; the principles given below will be extensively treated later in appropriate places. In the meantime handbook revisions or supplementary statements will be issued as rapidly as possible.

Since the statement of an average per capita dose for the whole population does not directly influence the substance of the NCRP Handbooks, no further statements regarding such a number will be made at this time. In any discussion of the MPD it is impractical to take into consideration the dose from natural background and medical or dental procedures.

The changes in the accumulated MPD are not the result of positive evidence of damage due to use of the earlier permissible dose levels, but rather are based on the desire to bring the MPD into accord with the trends of scientific opinion; it is recognized that there are still many uncertainties in the available data and information. Consideration has also been given to the probability of a large future increase in radiation uses. In spite of the trends, it is believed that the risk involved in delaying the activation of these recommendations is very small if not negligible. Conditions in existing installations should be modified to meet the new recommendations as soon as practicable, and the new MPD limits should be used in the design and planning of future apparatus and installations. Because of the impact of these changes and the time required to modify existing equipment and installations, it is recommended on the basis of present knowledge that a conversion period of not more than 5 years from January, 1957 be adopted within which time all necessary modifications should be completed.

The basic rules and the operational guides outlined below are intended to be in general conformity with the philosophy expressed in the 1953 statements of the ICRP, as revised in April, 1956 and March, 1958.

GUIDES FOR THE PREPARATION OF NCRP RECOMMENDATIONS

It is agreed that we should make clear distinction between basic MPD *rules* or requirements, and operational or administrative *guides* to be used according to the special requirements in any particular situation. Guides have the distinct value of retaining some reasonable degree of uniformity in the interpretation of the basic rules.

The risk to the individual is not precisely determinable but, however small, it is believed not to be zero. Even if the injury should prove to be proportional to the amount of radiation the individual receives, to the best of our present knowledge, the new permissible levels are thought not to constitute an unacceptable risk. Since the new rules are designed to limit the potential hazards to the individual and to the reproductive cells, it is therefore necessary to control the radiation dose to the population as a whole, as well as to the individual. For this reason, maximum permissible doses are set for the small percentage of the whole population who may be occupationally exposed, in order that they not be involved in risks greater than are normally accepted in industry. Also radiation workers represent a somewhat selected group in that individuals presumably of the greatest susceptibility (i.e., infants and children) are not included. However, for the persons located immediately outside of controlled areas but who may be exposed to radiation originating in controlled areas, the permissible level is adjusted downward from that in the controlled area because the number of such persons may not be negligible. With this downward adjustment, the risk to the *individual* is negligible so that small transient deviations from the prescribed levels are unimportant.

3-14 EXPOSURE STANDARDS AND PROTECTION REGULATIONS

Controls of radiation exposure should be adequate to provide reasonable assurance that recommended levels of maximum permissible dose shall not be exceeded. In addition, the NCRP reemphasizes its long-standing philosophy that radiation exposures from whatever sources should be as low as practical.

DEFINITIONS

For the purposes of these guides, the following definitions are given:

Controlled area. A defined area in which the occupational exposure of personnel to radiation or to radioactive material is under the supervision of an individual in charge of radiation protection. (This implies that a controlled area is one that requires control of access, occupancy, and working conditions for radiation protection purposes.)

Workload. The output of a radiation machine or a radioactive source integrated over a suitable time and expressed in appropriate units.

Occupancy factor. The factor by which the workload should be multiplied to correct for the degree or type of occupancy of the area in question.

RBE dose. RBE stands for relative biological effectiveness. An RBE dose is the dose measured in rems. (This is discussed in the report of the International Commission on Radiological Units and Measurements, 1956, NBS Handbook 62, p. 7.)

BASIC RULES

1. Accumulated Dose (Radiation Workers).

A. External exposure to critical organs.

Whole body, head and trunk, active blood-forming organs, or gonads: The maximum permissible dose (MPD), to the most critical organs, accumulated at any age, shall not exceed 5 rems multiplied by the number of years beyond age 18, and the dose in any 13 consecutive weeks shall not exceed 3 rems.

Thus the accumulated $MPD = 5 (N - 18)$ rems, where N is the age in years and is greater than 18.

COMMENT: This applies to radiation of sufficient penetrating power to affect a significant fraction of the critical tissue. (This will be enlarged upon in the revision of H59.)

B. External exposure to other than critical organs.

Skin of whole body: $MPD = 10 (N - 18)$ rems, and the dose in any 13 consecutive weeks shall not exceed 6 rems.

COMMENT: This rule applies to radiation of low penetrating power. See figure 2, H59.

Lens of the eyes: The dose to the lens of the eyes shall be limited by the dose to the head and trunk (A, above).

Hands and forearms, feet and ankles: $MPD = 75$ rems/year, and the dose in any 13 consecutive weeks shall not exceed 25 rems.

C. Internal exposures.

The permissible levels from internal emitters will be consistent as far as possible with the age-proration principles above. Control of the internal dose will be achieved by limiting the body burden of radioisotopes. This will generally be accomplished by control of the average concentration of radioactive materials in the air, water, or food taken into the body. Since it would be impractical to set different MPC values for air, water, and food for radiation workers as a function of age, the MPC values are selected in such a

manner that they conform to the above-stated limits when applied to the most restrictive case, viz., they are set to be applicable to radiation workers of age 18. Thus, the values are conservative and are applicable to radiation workers of any age (assuming there is no occupational exposure to radiation permitted at age less than 18). The factors entering into the calculations will be dealt with in detail in the forthcoming revision of Handbook 52.

The maximum permissible average concentrations of radionuclides in air and water are determined from biological data whenever such data are available, or are calculated on the basis of an averaged annual dose of 15 rems for most individual organs of the body, 30 rems when the critical organ is the thyroid or skin, and 5 rems when the gonads or the whole body is the critical organ. For bone seekers the maximum permissible limit is based on the distribution of the deposit, the RBE, and a comparison of the energy release in the bone with the energy release delivered by a maximum permissible body burden of 0.1 $\mu\text{g Ra}^{226}$ plus daughters.

2. *Emergency Dose (Radiation Workers).*

An accidental or emergency dose of 25 rems to the whole body or a major portion thereof, occurring only once in the lifetime of the person, need not be included in the determination of the radiation exposure status of that person (see p. 69, H59).

3. *Medical Dose (Radiation Workers).*

Radiation exposures resulting from necessary medical and dental procedures need not be included in the determination of the radiation exposure status of the person concerned.

4. *Dose to Persons Outside of Controlled Areas.*

The radiation or radioactive material outside a controlled area, attributable to normal operations within the controlled area, shall be such that it is improbable that any individual will receive a dose of more than 0.5 rem in any 1 year from external radiation.

The maximum permissible average body burden of radionuclides in persons outside of the controlled area and attributable to the operations within the controlled area shall not exceed one-tenth of that for radiation workers. This will normally entail control of the average concentrations in air or water at the point of intake, or rate of intake to the body in foodstuffs, to levels not exceeding one-tenth of the maximum permissible concentrations allowed in air, water, and foodstuffs for occupational exposure. The body burden and concentrations of radionuclides may be averaged over periods up to 1 year.

The maximum permissible dose and the maximum permissible concentrations of radionuclides as recommended above are primarily for the purpose of keeping the average dose to the whole population as low as reasonably possible, and not because of specific injury to the individual.

COMMENT: Occupancy-factor guides will be needed by several of the subcommittees. It will be important that these do not differ markedly between different handbooks. The Executive Committee will endeavor to establish a set of uniform occupancy-factor guides.

OPERATIONAL AND ADMINISTRATIVE GUIDES

5. The maximum dose of 12 rems in any 1 year as governed by the 13 week limitation, should be allowed only when adequate past and current exposure records exist. The allowance of a dose of 12 rems in any 1 year should not be encouraged as a part of routine operations; it should be regarded as an allowable but unusual condition. The records of previous exposures must show that the addition of such a dose will not cause the individual to exceed his age-prorated allowance.

3-16 EXPOSURE STANDARDS AND PROTECTION REGULATIONS

6. The full 3-rem dose should not be allowed to be taken within a short time interval under routine or ordinary circumstances (however, see paragraph 2 on Emergency Dose above). Desirably, it should be distributed in time as uniformly as possible and in any case *the dose* should not be greater than 3 rems in any 13 consecutive weeks. When the individual is not personally monitored and/or personal exposure records are not maintained, the exposure of 12 rems in a year should not be allowed; the yearly allowance under these circumstances should be 5 rems, provided area surveys indicate an adequate margin of safety.

7. When any person accepts employment in radiation work, it shall be assumed that he has received his age-prorated dose up to that time unless (1) satisfactory records from prior radiation employment show the contrary, or (2) it can be satisfactorily demonstrated that he has not been employed in radiation work. This is not to imply that such an individual should be expected to routinely accept exposures at radiation levels approaching the yearly maximum of 12 rems up to the time he reaches his age-prorated limit. Applications of these principles will serve to minimize abuse.

8. The new MPD standards stated above are not intended to be applied retroactively to individuals exposed under previously accepted standards.

9. It is implicit in the establishment of the basic protection rules that at present it is neither possible nor prudent to administer a suitably safe radiation protection plan on the basis of yearly monitoring only. It is also implicit that at the low permissible dose levels now being recommended, there is fairly wide latitude in the rate of delivery of this dose to an individual so long as the dose remains within the age-prorated limits specified above. In spite of a lack of clear evidence of harm due to irradiation at dose rates in excess of some specified level, it is prudent to set some reasonable upper limit to the rate at which an occupational exposure may be delivered. Therefore, it has been agreed that the dose to a radiation worker should not exceed 3 rems in any 13 consecutive weeks.

10. The latitude that may appropriately be applied in the operational and administrative control of occupational exposure will be dictated by two major factors (a) the type of risk involved and the likelihood of the occurrence of over-exposures and (b) the monitoring methods, equipment, and the dose recording procedures available to the radiation users. Where the hazards are minimal and not likely to change from day to day or where there are auxiliary controls to insure that the 13-week limitation will not be exceeded, the integration may be carried out over periods up to 3 months. Where the hazards are significant and where the exposure experience indicates unpredictability as to exposure levels, the doses should be determined more frequently, such as weekly, daily, hourly, or oftener, as may be required to limit the exposure to permissible values.

11. For the vast majority of installations (medical and industrial), operation is more or less routine and reasonably predictable and it may be expected that their monitoring procedures will be minimal. For such installations the protection design should be adequate to insure that over-exposures will not occur—otherwise frequent sampling tests should be specified. Where film badges are used for monitoring, it is preferable that they be worn for 4 weeks or longer, since otherwise the inaccuracy of the readings may unduly prejudice the radiation history of the individual. Where operations are not routine or are subject to unpredictable variations that may be hazardous, self-reading pocket dosimeters, pocket chambers, or other such devices should also be worn and should be read daily or more often as circumstances dictate.

12. Except for planning, convenience of calculation, design, or administrative guides, the NCRP will discontinue the use of a weekly MPD or MPC.

13. The Committee has deliberately omitted the discussion of future exposure forfeiture for exposures exceeding the MPD on the grounds that any such statements might lend encouragement to the unnecessary use of forfeiture provisions.

NBS Handbook 61—**Regulation of Radiation Exposure by Legislative Means**
(1955)

Recognizing the trend on both a national and state level for the development of radiation-protection regulations, the NCRP, in 1953, undertook to review and publish its philosophy regarding such regulations, together with a pattern for a suitable regulation and a complete legislative act that might be used where a state wished to set up a separate radiation-control agency.

Perhaps the outstanding need felt by the committee was that whatever regulations might exist at the Federal level, or the level of the various states, the need for uniformity was of highest urgency and importance. It is essential that there be no serious conflict between the regulations of the different states. To have it otherwise not only would be to invite confusion in interstate commerce but would also make it extremely difficult to manufacture radiation devices suitable for distribution anywhere in the country.

The committee recognizes the need for controlling radiation exposure but is not fully convinced that this can be best accomplished by legislation and regulations under all situations. It feels strongly that, whether or not there are regulations, strong educational efforts should be made to ensure that radiation users have adequate knowledge of the potential hazard.

Another item of importance has to do with the question of whether radiation regulations and legislation are enforceable. The committee felt that if they are developed in too great detail, or with too little understanding of the problem, it is highly conceivable that unenforceable regulations might be developed. This would quickly be learned by the users and there would develop a tendency to by-pass those parts that were unsatisfactory, in the knowledge that perhaps none of them would be enforced. Included in this problem of enforcement is the ability of the enforcing agency to do a proper policing job. If this cannot be done, the whole plan could easily fall into disrepute.

Another point of philosophy has to do with the question of licensing vs. registration. Licensing implies advance approval. If it is to be effective, the committee felt that this would constitute an almost unbearable burden on most agencies presently existing in this country. On the other hand, registration involving mainly notification to the control agency of the existence of sources would give the agency the knowledge of where to look for trouble. These sources could then be checked according to the size and the ability of the enforcing agency.

Another section deals with the constitution of the radiation-regulation body. This gives some indication as to the kind of competence that should be had within the control agency. Included are recommendations regarding the development and the use of advisory services on the part of highly trained experts in the radiation-protection field.

If the state decides it would need legislative powers in the matter of radiation protection, it has two kinds of acts available to it. One is known as an objectivity act which spells out the primary objectives that it seeks. The other alternative would be a specificity act which contains many details, technical data, exposure limits, etc. Since these very details are the ones that are most likely to change from time to time, it is recommended that no act include any such material.

3-18 EXPOSURE STANDARDS AND PROTECTION REGULATIONS

Another section includes a number of the more essential items that should be included in radiation-control acts or regulations. These cover such questions as policy, scope, definitions, exemptions, prohibitions, registrations, the development of the advisory board, and general matters of administration.

In Appendix A of NBS Handbook 61, there is given in detail a suggested **State Radiation Protection Act**, which covers matters mentioned above, together with more specific details, largely of a legal nature. It is doubted whether any state would adopt the entire act as proposed; in fact, it is recommended that a state not adopt any more of the act than it can properly implement.

Appendix B of NBS Handbook 61 gives a suggested set of uniform regulations containing objectives that may be adopted by the state. Here again, it is not believed that too many states would adopt these regulations in the exact and precise form. However, it would be essential to adopt the same primary technical considerations that were outlined in the report. At the time of writing, there is some indication that several states, such as Pennsylvania and Texas, are patterning their regulations in close accordance with those suggested here.

Important excerpts from NBS Handbook 61 follow:

1. SCOPE

It is the purpose of these regulations to state such requirements as shall be applied in the use of all radiation, radiation machines, and radioactive materials to insure the maximum safety to all persons at, or in the vicinity of, the place of use, storage, or disposal thereof. These regulations are intended to be consistent with the best use of radiation machines and radioactive materials.

2. APPLICATION

a. All radiation machines and radioactive materials shall be manufactured, used, stored, handled, transported, or disposed of in such manner that no person shall receive an excessive radiation dose therefrom. Except as exempted by the provisions of section 4 of these regulations, the manufacture, use, storage, handling, transportation, or disposal of radiation machines and radioactive materials shall be subject to the specific regulations provided below.

b. For the purposes of these regulations, radiation machines and radioactive materials used by, or in the possession of, an employee within the scope of his duties shall be considered to be in the possession of the employer.

3. DEFINITIONS

For the purposes of these regulations, the following definitions shall apply:

Absorbed dose of any radiation is the amount of energy imparted to matter by ionizing particles per unit mass of irradiated material at the place of interest.

Adult is a person of age 18 or more. This age limitation is for radiation-protection purposes only and bears no relationship to age limits for social, political, or other legal considerations (Sec. 6, NBS Handbook 59).

Agency is that governmental agency that is given the responsibility for administering these regulations.

Body burden is the amount of radioactive material in the body at the time of interest (Sec. D, NBS Handbook 52).

Critical organ is that part of the body that is most susceptible to radiation damage under the specific conditions considered.

Excessive radiation dose is a dose of radiation in excess of the maximum permissible dose. (Depending upon the degree of excess, these conditions represent serious hazard only when they are maintained or repeated frequently over long periods of time.)

Harmful effect is any body injury, disease, or impairment, except where such condition is transitory, infrequent, or of short duration, and does not endanger persons so affected.

Installation is the area of radiation hazard under the administrative control of the person or organization possessing the source of radiation.

Maximum permissible dose is a dose of radiation that, in the light of present knowledge, is not expected to cause appreciable bodily injury to a person at any time during his lifetime (Sec. 4.3, NBS Handbook 59).

Personnel monitoring is the determination of the radiation dose received by a person during a specified period (NBS Handbook 51).

Population group is a civil population (usually more than 100,000 persons) living in geographic proximity and generally dependent upon the same sources of food and water.

Qualified expert is a person fitted by training and experience to perform dependable radiation surveys, to oversee radiation monitoring, and to estimate the degree of radiation hazard. If the ability of a qualified expert is questioned, the Agency shall be the judge of his qualifications, in regard to which it may consider the testimony of other persons whom it deems expert.

Rad is the unit of absorbed dose and is equal to 100 ergs per gram. It is a measure of the energy imparted to matter by ionizing particles per unit mass of irradiated material at the place of interest (International Commission on Radiological Units, 1953). (See *Am. J. Roentgenol. Radium Therapy Nuclear Med.*, vol. 71, p. 139, 1954; *Radiology*, vol. 62, p. 106.)

Radiation is gamma rays and X-rays, alpha and beta particles, high-speed electrons, neutrons, protons, and other nuclear particles; but not sound or radio waves, or visible, infrared, or ultraviolet light.

Radiation hazard is any condition that might result in the exposure of persons to radiation in excess of the maximum permissible dose.

Radiation machine is any device that produces radiation when the associated control devices are operated.

Radioactive material is any material, solid, liquid, or gas, that emits radiation spontaneously.

Relative Biological Effectiveness (RBE) is the biological effectiveness of one type and energy of radiation, relative to that of lightly filtered X-rays generated at potentials of 200 to 300 kilovolts, for the particular biological system and biological effect, and for the conditions under which the radiation is received (Sec. 3.6, NBS Handbook 59).

Rem is the quantity of any radiation such that the energy imparted to a biological system (cell, tissue, organ, or organism) per gram of living matter by the ionizing particles present in the region of interest, has the same biological effectiveness as an absorbed dose of 1 rad from lightly filtered X-rays generated at potentials of 200 to 300 kilovolts. A dose in rems is equal to the dose in rads multiplied by the appropriate RBE (Sec. 4.8, NBS Handbook 59).

Sealed source is a quantity of radioactive material so enclosed as to prevent the escape of any radioactive material, but at the same time permitting radiation to come out for use (NBS Handbook 54).

Survey is the evaluation of radiation near a source by, or under the supervision of, a qualified expert (NBS Handbook 51).

Other scientific and technical terms not herein specifically defined shall be used in accordance with the definitions in (1) recommendations of the National Committee

3-20 EXPOSURE STANDARDS AND PROTECTION REGULATIONS

on Radiation Protection as published in Handbooks of the National Bureau of Standards, or (2) National Research Council Glossary of Terms in Nuclear Science and Technology (Published by American Society of Mechanical Engineers, 29 West 39th Street, New York City, New York, 1953), with preference being in the order given above.

4. EXEMPTIONS

a. These regulations shall not apply to the following materials, machines, or conditions:

(1) Natural radioactive materials of an equivalent specific radioactivity not exceeding that of natural potassium.

(2) Radioactive material in such quantity that if the entire amount were taken internally, continuously, or at one time by a person, no harmful effect would be likely to result. Listings of the upper limits of quantities of radioactive materials that shall be exempt from these regulations and from registration are given in section 15.c and Table 3-2 of these regulations. These limits apply only for radioactive material not contained in sealed sources.

(3) Radioactive materials in sealed sources in total quantities not exceeding 1 millicurie for a given installation.

(4) Timepieces, instruments, novelties, or devices containing self-luminous elements, except during manufacture or repair of the self-luminous elements themselves.

(5) Electrical equipment that is primarily not intended to produce radiation and that, by nature of design, does not produce radiation at the point of nearest approach at a weekly rate higher than one tenth ($1/10$) the appropriate permissible dose for any critical organ exposed. The production testing or production servicing of such equipment shall not be exempt.

(6) Radiation machines not being used in such manner as to produce radiation.

(7) Any radioactive material being transported on vessels, aircraft, railroad cars, or motor vehicles in conformity with the regulations adopted by any agency having jurisdiction over safety during transportation.

(8) The Agency may exempt radiation machines or radioactive materials known to be without hazard, and shall authorize the labeling of the ones that it does exempt.

b. Nothing in these regulations shall be construed to limit the kind and amount of radiation that may be intentionally applied to a person for diagnostic or therapeutic purposes by, or under the direction of, a physician or dentist.

c. (Applications for exemptions to conform to general State practices.)

5. STANDARDS

Recommendations of the National Committee on Radiation Protection as published in Handbooks of the National Bureau of Standards shall be used as guides or standards or as a basis for calculations to obtain or maintain safe working conditions within the meaning of the regulations herein, but shall not be considered in whole or in part as a portion of these regulations unless specifically so stated.

Alternate: Recommendations of the National Committee on Radiation Protection as published in National Bureau of Standards Handbooks 42, 48, 49, 51, 52, 53, 54, 55, 56, 58, 59, and 60 shall be used as guides or standards or as a basis for calculations to obtain or maintain safe working conditions within the meaning of the regulations herein. (If this Alternate is used, appropriate parts of sections 3, 7, and 15 of these regulations may be omitted.)

6. REGISTRATION

a. Any person using or operating any radiation machine, or storing, manufacturing, using, or handling any radioactive material, shall notify the Agency of the fact in writing within 30 days following the commencement thereof. Said notice shall state the location, nature, and scope of such operation, use, or storage, and shall be reviewed and if necessary brought up to date annually thereafter.

b. The notification in paragraph (a) above shall include an estimate of any further accession of radiation machines or radioactive material expected during the ensuing year. Any accession in excess of the estimate shall be registered promptly.

c. Acknowledgment of registration shall not imply the approval by the Agency of the manufacture, storage, use, or operation described in the registration, but shall merely indicate that the Agency has a record of the locations and establishments where radiations are used.

7. MAXIMUM PERMISSIBLE DOSE

a. The exposure of persons to radiation shall always be kept to the lowest practicable level.

b. When the source of radiation is outside the body, the maximum permissible dose rate shall not exceed those values specified in section 15.d of these regulations (NBS Handbook 59).

c. Quantities of radioactive material on the surface of the body, or on clothing worn by the person, shall not exceed those that will result in average dose rates to any portion of the body greater than the applicable permissible value specified in section 15.d of these regulations. (The skin will generally be the critical organ for sufficiently small sources of radiation. For such cases, the area over which the dose is averaged shall be of the order of 1 square centimeter.) Wounds, cuts, or abrasions of the skin involving contamination shall be given immediate medical attention for the removal of such contamination (NBS Handbook 48).

d. The maximum permissible dose for an individual shall be considered to include all doses, from internal and external sources, from all types and energies of radiation, whether delivered simultaneously or successively, to the region of interest, during the period of measurement.

e. Radiation dose to the tissues of the body from radioactive materials within the body shall be controlled by limiting the average rates at which radioactive materials are taken into the body either by inhalation or by ingestion. Where such intake results from the occurrence of a radioisotope in air or water, the average concentration of the radioisotope in the air or water used by the individual shall not exceed the maximum permissible concentration specified in Table 3-3 of these regulations (NBS Handbook 52).

f. The determination of the dose received by persons and degree of hazard present in all places to which these regulations apply shall be guided by nationally recognized standards such as (1) the recommendations of the National Committee on Radiation Protection as published in Handbooks of the National Bureau of Standards, and (2) Safety Standards of the American Standards Association.

g. The radiation dose to any population group shall be limited to one-tenth ($\frac{1}{10}$) the maximum permissible amounts stated in section 15 of these regulations (International Commission on Radiological Protection, *Brit. J. Radiol., Suppl.* 6).

8. PERSONNEL MONITORING: AREA RADIATION SURVEYS

a. All accessible areas in the vicinity of radiation-producing sources shall be surveyed by, or under the direction of, a qualified expert using suitable instruments and methods

3-22 EXPOSURE STANDARDS AND PROTECTION REGULATIONS

for measuring radiation, to determine the maximum levels of radiation to which persons may be exposed. For protection purposes these measurements shall be reduced to weekly doses taking into consideration the amount of time the radiation is being produced, the work week, and the fraction of the week that any person might be exposed to the radiation.

b. In lieu of an actual survey a written statement made by a qualified expert based on his analysis of the situation shall be acceptable as evidence of the absence of radiation hazard in a given area.

c. Personnel monitoring shall be required for each individual for whom there is any reasonable possibility of receiving a weekly dose of all radiations exceeding one-quarter ($\frac{1}{4}$) of the maximum permissible amounts specified in sections 7.b through 7.f of these regulations taking into consideration the use of protective gloves, aprons, or other radiation-limiting devices; except that, continuation of personnel monitoring shall not be required if the average dose over a period of 8 weeks proves to be less than one-half ($\frac{1}{2}$) the maximum permissible amounts specified in sections 7.b through 7.f of these regulations. If the specified operating conditions are changed, a new monitoring test over an 8-week period shall be made.

d. Routine monitoring of persons occupationally exposed to radiation from radiation machines shall not be required when all of the following conditions are met:

(1) A qualified expert has specified the operating conditions under which there is no reasonable chance that any person will be exposed to more than one-quarter ($\frac{1}{4}$) the maximum permissible dose.

(2) The operating conditions in (1) above are made known to all persons who may be occupationally exposed to the radiation.

(3) The installation continues to operate only under the specified conditions.

e. Regularly scheduled monitoring of the air within the installation for radiation or radioactive content shall be required when there is any reasonable possibility that the average levels of activity therein may exceed one-quarter ($\frac{1}{4}$) the amount specified in Table 3-3 of these regulations. Measurements averaged over a maximum period of 13 weeks shall be permissible, but in any case there shall be sufficient monitoring to insure that no radiation hazard exists.

9. RADIATION-EXPOSURE RECORDS AND REPORTS

a. Records of all measurements required shall be kept available for inspection by the Agency or its representative upon demand. A state may wish to specify further the length of time that such records should be kept. It is suggested that this be for the lifetime of the individual plus 2 years, or until he will have reached 70 years of age. Personnel-monitoring records shall include the Social Security numbers of the workers concerned as an aid in keeping track of an individual's total exposure.

b. Records of the amount, kind, and disposition of radioactive materials purposefully removed from the installation shall be maintained and available for inspection by the Agency or its representative upon demand.

c. Upon termination of employment of a person, the Agency shall, upon request, be supplied with a summary statement of that person's average radiation dose. (The estimated maximum dose shall be stated if no personnel monitoring has been carried out.) This record shall include statements of any circumstances wherein the dose to the employee, from any source of radiation, exceeded those specified in these regulations. (The purpose of this requirement is to establish a central point for the maintenance of radiation-exposure records.)

d. When it is known or believed that an accidental dose to a person in the installation may have exceeded 5 times the amount permitted by applicable portions of sections 7.b

through 7.f of these regulations, all facts relative to the occurrence shall be reported in detail to the Agency within 7 days of the discovery thereof, and a copy of the report shall be put in that person's personnel file. The cause of the overexposure shall immediately be sought out and corrected.

10. RESPONSIBILITY

a. All work performed in an installation where radiation may be present shall be under the direction of a person responsible for the radiation safety therein. His name shall be reported to the Agency.

b. The person in charge of the radiation safety in an installation shall have the following responsibilities:

(1) He shall inform himself of the hazards attendant upon the presence of radiation in the installation and, if necessary to this end, obtain the services of a qualified expert.

(2) He shall provide, or cause to be provided, any necessary instruction concerning the attendant radiation hazards and safe working practices, to all employees whose duties necessitate the handling of radioactive material or the operation of any machines that produce radiation in amount that leads to hazard, and to all other employees who are not regularly employed at such work but who may occasionally be exposed to radiation.

(3) He shall insure beyond reasonable doubt that all persons working with radiation machines or radioactive materials, and all authorized visitors to areas where radiation may be present, are properly and adequately instructed in the use of all necessary safeguards and procedures, and are supplied with such auxiliary devices as may be necessary for safety.

(4) He shall insure beyond reasonable doubt that no radioactive material (including that in patients, animals, and equipment) is allowed to leave the jurisdiction of the radiation user under circumstances that may subject other persons to radiation in amounts in excess of those indicated in sections 7.b through 7.f of these regulations.

(5) He shall insure beyond reasonable doubt that any area, inside or outside the installation, normally occupied by adults not primarily engaged in radiation or associated work, cannot be subjected to radiation levels exceeding the maximum permissible amounts indicated in sections 7.b through 7.f of these regulations.

(6) He shall insure beyond reasonable doubt that any area, inside or outside the installation, that may be habitually occupied by persons under 45 years of age and not engaged in radiation work, cannot be subjected to radiation levels exceeding one-tenth ($\frac{1}{10}$) the maximum permissible amounts indicated in sections 7.b through 7.f of these regulations, except that such exposure may be averaged over 1 year. Normal occupational exposures of pregnant women should be considered as acceptable risks.

(7) He shall notify the building superintendent or other appropriate official of the existence of any areas not normally occupied but in which hazardous radiation exposure may take place; e.g., an air-conditioning equipment room beneath X-ray installation.

(8) He shall notify the building superintendent or other appropriate official of the existence of any conditions or situations that, while not normally considered a radiation hazard, may become a hazard under special or unusual circumstances; e.g., entrapment of waste in a drainage line.

(9) He shall, by means of appropriate surveying or monitoring procedures, insure that radioactivity discharged to the atmosphere, at any point where persons may breathe the air, shall be maintained at an average concentration of radioactivity below the maximum permissible levels indicated in section 7 of these regulations.

3-24 EXPOSURE STANDARDS AND PROTECTION REGULATIONS

c. Every employee and authorized visitor shall be responsible for using such safety devices as are furnished for his protection and for carrying out all radiation-safety rules that concern or affect his conduct.

11. STORAGE OF RADIOACTIVE MATERIALS

a. Radioactive materials shall be stored or kept in such a manner as to insure that the dose rate therefrom shall not exceed the appropriate limits specified in section 7 of these regulations.

b. Vaults or rooms in which radioactive materials are stored shall be so located and/or constructed that no person shall be exposed to radiation therefrom in excess of the appropriate limits set forth in sections 7.b through 7.f of these regulations.

c. Radioactive materials in a workroom or other location where persons are regularly or frequently present shall be enclosed in containers of such thickness, material, and construction, or otherwise shielded in such manner, that no person will be exposed to radiation in amounts greater than those indicated in sections 7.b through 7.f of these regulations.

d. Vaults or rooms used for storing materials that may emit radioactive gases shall be suitably ventilated in such a manner that the gases do not constitute a radiation hazard.

e. When there is any possibility that chemical, radiation, or other action might weaken or rupture the container of radioactive material sufficiently to cause leakage therefrom, the container shall be provided with a suitable secondary tray or catchment adequate to retain the entire amount of radioactive material.

f. Each container of radioactive material in storage shall, in addition to the standard radiation-hazard symbol (see section 13.c of these regulations), be labeled in such manner that the kind and quantity of material, the date of measurement, and the name of the person responsible for the material can be easily and quickly determined.

g. Storage containers for radioactive material in excess of 1 curie shall be designed to be resistant to fire and earthquake damage, and to maintain reasonable temperatures. Containers shall be structurally sound over the period of intended use with due regard to corrosion, radiation, and temperature effects that may develop.

h. Suitable provision shall be made to minimize the hazard to emergency workers in the event of fire and in situations where earthquake, flood, and windstorm potentials exist.

12. RADIOACTIVE-CONTAMINATION CONTROL

a. All work with radioactive materials shall be carried out under such conditions as to minimize the possibility of any contamination that would result in any person's being subjected to radiation levels exceeding those specified in section 7 of these regulations.

b. Where the nature of the work is such that a person or his clothing may become contaminated to such degree as to present a hazard, both shall be suitably monitored. Any contamination leading to doses in excess of the values specified in section 7 of these regulations shall be removed from the contaminated person before that person is permitted to leave the work area. Clothing or other material having contamination in excess of the amounts indicated in section 7 of these regulations shall not be taken from the work area or released to public laundries or cleaners.

c. Under conditions in which the Agency considers it advisable it may devise or approve a suitable pattern of work rules applicable to individual users. These may vary from one user to another.

d. Every person using radioactive materials not enclosed in a sealed source shall have on hand or immediately available an instrument or instruments suitable for detecting and measuring contamination in accordance with the requirements of this section. These instruments shall be maintained in proper calibration. Under special circumstances, the Agency may require the same or similar instrumentation for users of radioactive materials in sealed sources.

e. Any accidental release of radioactive material beyond the control or jurisdiction of the installation shall be reported to the Agency immediately.

13. RADIATION INFORMATION LABELING

a. All radiation machines shall be clearly labeled as follows:

"CAUTION—X-RAYS"

This equipment produces X-rays when energized."

(Labels as required under the Federal Food, Drug, and Cosmetic Act may be substituted for the above label.)

b. All radioactive material not in process or in possession of the user shall be clearly labeled as follows:

(1) Containers for sealed sources of external hazard only:

"CAUTION—RADIATION"

(Where a time limit is specified, it should be posted.)

(2) Radioactive material in loose bulk or unsealed containers—internal hazards primarily:

"DANGER—RADIOACTIVE MATERIAL"

The material contained herein should not be allowed to enter the body either by inhalation, ingestion, or through wounds in the skin."

(Labels as required under the Atomic Energy Act may be substituted, where appropriate, for the above labels.)

c. The standard symbol for designating any radiation hazard shall be:



The standard color specification shall be a background of yellow with lettering and distinctive symbol in purple (magenta). The use of this symbol for any other purpose is expressly prohibited. The symbol and lettering shall be as large as practical, consistent with size of the equipment or material.

d. All radioactivity containers, storage areas, work areas, or other normally occupied areas where a radiation hazard may exist shall be posted with accepted radiation-hazard labels, except where such labels may be a source of disturbance to patients undergoing radiation treatment.

e. Any areas where a radiation hazard may exist on a frequent or infrequent basis, but which are not readily accessible and are so situated as to be occupied only under

infrequent and special circumstances, shall be posted with accepted radiation-hazard labels.

f. All areas that are readily accessible but not normally occupied, and where a radiation hazard may exist on a frequent or infrequent basis, shall be suitably fenced off and posted with the accepted radiation-hazard label.

g. All radiation-hazard labels posted when a radiation hazard existed shall be removed when the hazard is no longer present.

14. DISPOSAL OF RADIOACTIVE WASTES

(Note: It is impractical or impossible at the present time to formulate detailed waste-disposal regulations that will provide adequate safety under all conditions without being unnecessarily restrictive under most conditions. Not only do the relevant factors vary with each locality, but virtually every radioisotope presents a different problem. A few broad rules applicable to cases of most common interest can be stated, but in general the procedures must be adjusted, in the light of the general guides specified in section 7 of these regulations, to take into account the particular circumstances involved.)

The disposal of radioactive waste, by the nature of the problem, develops situations different from those encountered in the normal handling of radioactive material under controlled conditions. Once a disposal event has taken place all further control over the material usually is completely out of the hands of the disposer; an irreversible train of events will have begun. It is therefore essential that special care and consideration be exercised before any disposal operation is commenced.)

a. Users of radioactive materials shall release these materials only in such manner that the radioactive material discharged, in combination with that discharged by other users, will not cause contamination of the environment that may result in a person or persons receiving an excessive radiation dose. If several users are discharging radioactive wastes to the same environment, they shall, upon being notified of the fact, cooperate in limiting the release and shall file with the Agency a statement of their agreed pro-rata releases. If this is not done within a reasonable time the agency arbitrarily may assign quotas to them severally.

b. Users who release radioactive material may take reasonable advantage of the environmental factors (dilution, dispersion, etc.) to minimize the cost of disposal, provided they meet the performance requirements indicated in paragraph (a) above. Nothing in these regulations shall be construed as permitting release of materials that would be unlawful for other reasons.

c. Prior to and during design and construction of facilities for the handling and disposal of radioactive wastes, users of radioactive materials may obtain opinions from the Agency regarding the probability of meeting these regulations. However, the user shall remain responsible for meeting the performance standards related to radioactivity established by the Agency and shall allow representatives of the Agency to inspect and evaluate his methods of treatment and release.

d. For purposes of protection of *population groups*, the limits of radioactivity resulting from disposal of radioactive material shall be determined on the following basis:

(1) The average concentration of that isotope in air at points where it is commonly used by humans or in water at points of supply (exclusive of treatment, if any) prior to use by humans shall not exceed ten percent (10%) of the maximum permissible levels recommended in Table 3-3, section 15 of these regulations. Concentrations lasting only over a period of a few days may be allowed to exceed the values given in Table 3-3, provided the average concentration over any interval of one year does not exceed ten percent (10%) of these values.

(2) Average rates of radiation dose to persons from radioisotopes outside their bodies shall not exceed ten percent (10%) of the values specified in Rules I through IV, section 15.d of these regulations.

(3) Average concentrations in portions of public waterways not used as sources for human consumption shall be consistent with health, economic, and recreational uses of the water and the future plans for its use that are under consideration by responsible authorities.

(4) For any radioisotope where the effective half-life in the body is less than 60 days, the term "average" as used above shall mean the arithmetic mean of a series of determinations representative of plant operations and environmental conditions over any period of 13 consecutive weeks; for other radioisotopes this arithmetic mean shall be taken over a period of any 12 consecutive months.

(5) If the permissible average concentration of a mixture of radioisotopes in air or water depends almost entirely on the concentration of one of the radioisotopes involved, for routine practical estimates the contributions of the other radioisotopes in the mixture may be neglected. Where more than one isotope is important, the permissible concentration shall be arrived at by adding the exposures to be received from each significant component (page 42, appendix A, NBS Handbook 52).

e. Radioactive wastes may be disposed of by dumping or burial only in areas approved by the Agency for that purpose. Areas approved for this purpose shall be designed and operated so that they comply with the other provisions of these regulations. (NBS Handbook 58 and a forthcoming report of the NCRP Subcommittee on Waste Disposal and Decontamination, "Burial of Radioactive Wastes in Soil.")

15. TECHNICAL STANDARDS, GUIDES, AND GENERAL INFORMATION TO BE USED IN ACHIEVING THE REQUIREMENTS OF THESE REGULATIONS

a. It will be considered that the data in this section are the best currently available to serve as guides for the fulfillment of the spirit and intent of the regulations.

b. Modifications in this section will be made whenever the Agency finds it necessary in order to conform to the best practice and new information made known through continuing research.

Table 3-2. Ranges of Limiting Quantities of Common Radionuclides

Group 1: 1 microcurie

Pb²¹⁰, Ra²²⁶, Ac²²⁷, Pu²³⁹, Am²⁴¹, Cm²⁴², Po²¹⁹, At²¹¹, U²³³

Group 2: 10 microcuries

Sc⁴⁶, Co⁶⁰, Sr⁹⁰, Ru¹⁰⁶, Ag¹⁰⁵, Te¹²⁹, I¹³¹, Cs¹³⁷, Ce¹⁴⁴, Eu¹⁵⁴, W¹⁸¹, Re¹⁸³, Ir¹⁹²

Group 3: 100 microcuries

P³², Cl³⁶, Ca⁴⁶, Sc⁴⁸, Sc⁴⁸, V⁴⁸, Fe⁵⁹, Zn⁶⁵, Ga⁷², As⁷⁶, Rb⁸⁶, Sr⁸⁹, Y⁹¹, Nb⁹⁵, Tc⁹⁶, Rh¹⁰⁵, Ag¹¹¹, Cd¹⁰⁹, Sn¹¹³, Te¹²⁷, Ba¹⁴⁰, La¹⁴⁰, Pr¹⁴³, Sm¹⁵¹, Ho¹⁶⁶, Tm¹⁷⁰, Lu¹⁷⁷, Ta¹⁸², Pt¹⁹¹, Pt¹⁹³, Au¹⁹⁸, Au¹⁹⁹, Tl²⁰⁰, Tl²⁰⁴, Pb²⁰³, Th²³⁴

Group 4: 1,000 microcuries

H³, Be⁷, C¹⁴, Na²⁴, S³⁵, K⁴², Cr⁵¹, Mn⁵⁶, Fe⁵⁵, Ni⁵⁹, Cu⁶⁴, Ge⁷¹, Mo⁹⁹, Pd¹⁰³, Pm¹⁴⁷, Ir¹⁹⁰, Au¹⁹⁶, Tl²⁰¹, Tl²⁰², natural uranium, natural thorium

Table 3-3. Maximum Permissible Amounts of Radioisotopes in Total Body and Maximum Permissible Concentrations in Air and Water

Group	Radioisotope	Column I * Microcuries per milliliter of air	Column II * Microcuries per milliliter of water	Column III Microcuries in total body
2	A ⁴¹	5×10^{-7}	5×10^{-4}	30
3	Ag ¹⁰⁶	1×10^{-5}	2	18
3	Ag ¹¹¹	3×10^{-5}	4	36
1	Am ²⁴¹	3×10^{-11}	1×10^{-4}	0.06
3	As ⁷⁶	2×10^{-5}	0.2	10
1	At ²¹¹	3×10^{-10}	2×10^{-5}	6×10^{-4}
3	Au ¹⁹⁸	1×10^{-7}	3×10^{-3}	10
3	Au ¹⁹⁹	2.5×10^{-7}	7×10^{-3}	28
3	Ba ¹⁴⁰ + La ¹⁴⁰	6×10^{-8}	2×10^{-3}	5
4	Be ⁷	4×10^{-5}	1	670
4	C ¹⁴	5×10^{-7}	3×10^{-3}	250
3	Ca ⁴⁵	3×10^{-8}	5×10^{-4}	65
3	Cd ¹⁰⁹ + Ag ^{109m}	7×10^{-8}	7×10^{-2}	40
2	Ce ¹⁴⁴ + Pr ¹⁴⁴	7×10^{-9}	4×10^{-2}	5
3	Cl ³⁶	4×10^{-7}	2×10^{-3}	200
1	Cm ²⁴²	2×10^{-10}	9×10^{-4}	0.05
2	Co ⁶⁰	1×10^{-6}	2×10^{-2}	3
4	Cr ⁵¹	8×10^{-6}	0.5	390
2	Cs ¹³⁷ + Ba ^{137m}	2×10^{-7}	1.5×10^{-3}	90
4	Cu ⁶⁴	6×10^{-6}	8×10^{-2}	150
2	Eu ¹⁵⁴	6×10^{-9}	3×10^{-2}	22
	F ¹⁸	1×10^{-4}	0.9	24
4	Fe ⁵⁵	6×10^{-7}	4×10^{-3}	1,000
3	Fe ⁵⁹	1.5×10^{-8}	1×10^{-4}	11
3	Ga ⁷²	3×10^{-6}	9	8
4	Ge ⁷¹	4×10^{-5}	9	67
4	H ³ (HTO or T ₂ O)	2×10^{-5}	0.2	10 ⁴
3	Ho ¹⁶⁶	3×10^{-6}	23	17
2	I ¹³¹	5×10^{-9}	3×10^{-5}	0.7
4	Ir ¹⁹⁰	7×10^{-7}	1×10^{-2}	21
2	Ir ¹⁹²	5×10^{-8}	9×10^{-4}	3.4
4	K ⁴²	2×10^{-6}	1×10^{-2}	20
3	La ¹⁴⁰	1×10^{-6}	1	24
3	Lu ¹⁷⁷	5×10^{-6}	24	78
4	Mn ⁵⁶	3×10^{-6}	0.15	2

Table 3-3. (Continued)

Group	Radioisotope	Column I * Microcuries per milliliter of air	Column II * Microcuries per milliliter of water	Column III Microcuries in total body
4	Mo ⁹⁹	2×10^{-3}	14	50
4	Na ²⁴	2×10^{-6}	8×10^{-3}	15
3	Nb ⁹⁵	4×10^{-7}	4×10^{-3}	90
4	Ni ⁵⁹	2×10^{-5}	0.25	39
3	P ³²	1×10^{-7}	2×10^{-4}	10
3	Pb ²⁰³	6.5×10^{-6}	0.1	57
4	Pd ¹⁰³ + Rh ¹⁰³	7×10^{-7}	1×10^{-2}	6
4	Pm ¹⁴⁷	2×10^{-7}	1	120
1	Po ²¹⁰ (sol.)	2×10^{-10}	3×10^{-6}	0.02
1	Po ²¹⁰ (insol.)	7×10^{-11}		7×10^{-3}
3	Pr ¹⁴³	7.5×10^{-7}	0.4	29
1	Pu ²³⁹ (sol.)	2×10^{-12}	1.5×10^{-6}	0.04
1	Pu ²³⁹ (insol.)	2×10^{-12}		0.008
1	Ra ²²⁶ + $\frac{1}{2}$ dr ^b	8×10^{-12}	4×10^{-8}	0.1
3	Rb ⁸⁶	4×10^{-7}	3×10^{-3}	60
2	Re ¹⁸³	8×10^{-6}	8×10^{-2}	35
3	Rh ¹⁰⁵	1×10^{-6}	1.5×10^{-2}	9
	Rn ²²² + dr ^b	^c 1×10^{-7}	2×10^{-6}	
2	Ru ¹⁰⁶ + Rh ¹⁰⁶	3×10^{-8}	0.1	4
4	S ³⁵	1×10^{-6}	5×10^{-3}	300
2	Sc ⁴⁶	7×10^{-8}	0.4	6
3	Sm ¹⁵¹	1×10^{-8}	0.2	420
3	Sn ¹¹³	6×10^{-7}	0.2	80
3	Sr ⁸⁹	2×10^{-8}	7×10^{-5}	2
2	Sr ⁹⁰ + Y ⁹⁰	2×10^{-10}	8×10^{-7}	1
3	Tc ⁹⁶	3×10^{-6}	3×10^{-2}	5
3	Te ¹²⁷	1×10^{-7}	3×10^{-2}	4
2	Te ¹²⁹	4×10^{-8}	1×10^{-2}	1.3
3	Th ²³⁴	6×10^{-7}	3	120
4	Th—natural (insol.)	3×10^{-11}		0.002
4	Th—natural	3×10^{-11}	4×10^{-7}	0.01
3	Tm ¹⁷⁰	5×10^{-8}	0.25	19
1	U ²³³ (sol.)	1×10^{-10}	1.5×10^{-4}	0.04
1	U ²³³ (insol.)	1.6×10^{-11}		0.008
4	U—natural (sol.)	1.7×10^{-11}	7×10^{-5}	0.02

Table 3-3. (Continued)

Group	Radioisotope	Column I ^a Microcuries per milliliter of air	Column II ^a Microcuries per milliliter of water	Column III Microcuries in total body
4	U—natural (insol.)	1.7×10^{-11}		0.009
3	Y ⁴⁸	1×10^{-6}	0.5	20
	Xe ¹³³	4×10^{-6}	4×10^{-3}	300
3	Xe ¹³⁵	2×10^{-6}	1×10^{-3}	100
3	Y ⁹¹	4×10^{-8}	0.2	15
3	Zn ⁶⁵	2×10^{-6}	6×10^{-2}	430
	All other beta or gamma emitters	1×10^{-9}	1×10^{-7}	
	All other alpha emitters	5×10^{-12}	1×10^{-7}	

^a The values given in columns I and II apply to continuous exposures for 24 hr a day. Where exposure is incurred only during a work day of 8 hr, the values in columns I and II may be multiplied by a factor of 3.

^b dr stands for daughter products.

^c The limit given in NBS Handbook 52 is 10^{-8} $\mu\text{C}/\text{ml}$. The new value, 10^{-7} $\mu\text{C}/\text{ml}$, was adopted by the International Commission on Radiological Protection, 1953, and is now acceptable in the United States.

c. *Maximum amounts of radioactive material permitted without registration.* Registration shall not be required for the possession or use of radioactive materials when the total quantities of one or more kinds of radioactive material in any one of the following groups, at any one time, is not exceeded:

- Group 1, 1 microcurie
- Group 2, 10 microcuries
- Group 3, 100 microcuries
- Group 4, 1,000 microcuries

Table 3-2 indicates the place of individual radioactive materials in the group. Any radioactive material not listed in Table 3-2 shall be considered as being in Group 2.

d. *Permissible dose from external sources of radiation.* The permissible doses from external sources of radiation are given under "Basic Rules," paragraphs 1A and 1B, given previously in this section on page 3-12.

e. *Exposure to sources of radiation within the body.* The permissible doses from sources of radiation within the body are given under "Basic Rules," paragraph 1C, given previously in this section on page 3-8. See also Table 3-3.

GOVERNMENT AND NONGOVERNMENT AGENCIES ESTABLISHING RADIATION STANDARDS AND REGULATIONS

(Other than the National Committee on Radiation Protection)

The development of standards in radiation protection has been proceeding through the exercise of interest by groups other than the National Committee on Radiation Protection. These groups have come into existence through the need for occupational health and safety standards generally, radiation protection being one part of their total scope.

American Standards Association

The **American Standards Association** was founded in 1918 by five engineering societies which were soon joined by three departments of the Federal government. It was formed primarily as a national clearinghouse for standards of all types—it is essentially a service organization, a federation of trade associations and technical societies. It functions as the machinery through the use of which standards or ideas for standards may be coordinated.

The American Standards Association considers standards to be a solution of a recurring difficulty—an agreement by authority, custom, or general consent to a rule or model to be followed whenever the difficulty arises. Through the use of ASA machinery, government and industry cooperate in the development of American Standards. The National Bureau of Standards, which also actively participated in the formation of the National Committee on Radiation Protection, has participated in the development of nearly half of all American Standards now in effect.

As a major result of this cooperation, government has begun requiring the use of American Standards as a basis of specifications rather than writing and requiring the use of its own specifications. The American Standards program is basically the development by industry of standards which it feels advantageous to use as uniform provisions for safety to life and for health.

Through the mechanics of ASA, safety standards have been developed for many types of industrial machinery and operations. About 160 **American Safety Standards** cover such items as accident-prevention signs, blower systems, building exits code, protective occupational clothing, coal mines, color code for marking physical hazards, compiling accident causes and work-injury experience data, electrical codes, exhaust systems, gas-mask canisters, head and eye protection, lighting codes, piping-identification systems, safety shoes, accident statistics, allowable concentrations of toxic dusts and gases, and industrial use of X-rays.

The ASA has stated that **Federal and state health and labor departments** use American Standard codes of permissible limits for 15 toxic substances widely used in industry (237). Other similar adoption of American Standards by government agencies is indicated in the statements that “176 municipalities have passed legal enactments based in whole or in part on the American Standard code covering the wiring of buildings (National Electrical Code, C1)” ; “The Safety Code for the Use, Care, and Protection of Abrasive Wheels (B7) is used by all American manufacturers of grinding wheels, and has been adopted either verbatim or in substance

3-32 EXPOSURE STANDARDS AND PROTECTION REGULATIONS

by all those states having regulations on the subject"; and "The New York City ordinances on the subject is a verbatim copy of the code" (The National Elevator Code A17.1-31).

At the request of the U.S. Department of Labor, Division of Labor Standards, the ASA initiated and approved on June 16, 1944, a war project looking into the development of an **American War Standard, Protection in the Use of X-ray Equipment**, designated **Code Z54.1**. The suggestion was also made that the National Bureau of Standards sponsor a peacetime project in this field under the regular procedure of ASA. The first draft of the War Standard was agreed upon at a meeting of the Z54 Committee on Nov. 9, 1944.

Part I of this standard was adopted in 1945 and set a tolerance dose at 0.1 r/day. The standard set up a classification of X-ray installations according to the degree of protection required, recommended that a qualified expert be consulted when planning a new installation or changes in an existing installation, recommended the wearing of film dosimeters, recommended regular monitoring for radiation, and suggested medical **supervision of X-ray workers'** health. The complete code was finished in 1946.

An **American Standard Safety Color Code** for Marking Physical Hazards and the Identification of Certain Equipment was approved Sept. 11, 1953, as Z-53.1-1953. This standard states that purple shall be the basic **radiation hazards color**. Yellow should be used in combination with purple for markers such as tags, labels, signs, and floor markers. Radiation as used in this code applies to radiation types such as alpha, beta, gamma, neutron, proton, deuteron, and meson. A partial list of suggestions for the application of the color designation is included. The colors are precisely defined in terms of CIE Chromaticity Coordinates and Reflectance Limits, Ideal Munsell Notation, and ISCC-NBS Color Designation.

The American Standards Association called a **General Conference on National Standardization in the Field of Nuclear Energy** on Dec. 8, 1955. It was unanimously voted that ASA establish a Planning Committee to study the matter and report back as soon as possible to the conference members.

After several meetings, in full committee and in several subcommittees, the Planning Committee recommended that "an ASA Standards Board" be set up in the field of nuclear energy, and that this **Nuclear Standards Board** be organized as soon as possible. It recommended that the initial membership of the NSB be essentially representative of the Planning Committee membership itself.

As a scope for the NSB, it was recommended that it cover the administration and planning of national standardization in the field of nuclear energy, with primary jurisdiction over work which does not fall under existing standards boards and secondary jurisdiction over related projects currently assigned to an existing standards board. It was understood that the technical work would be placed in the hands of existing sectional committees and existing committees of other organizations to the maximum extent possible. On June 1, 1956, the Standards Council met and voted to establish the NSB, after a letter ballot of the General Conference had recommended approval of the Planning Committee's suggestions. The June 1 meeting vote of the Standards Council was subsequently affirmed by a full letter ballot of the Council.

Program. The duties and responsibilities of standards boards are outlined in Section B4 of the ASA bylaws. ASA procedures for approving standards as "American Standard" are the Existing Standards Method, the Sectional Committee Method, and General Acceptance or Conference Method, which have as their common fundamental requirement that the ASA "**consensus**" principle must determine whether a particular set of specifications or standards can bear the "American Standard" designation.

Membership of the ASA Nuclear Standards Board includes representatives of the following:

Aircraft Industries Association
American Chemical Society
American Federation of Labor and Congress of Industrial Organizations
American Industrial Hygiene Association
American Institute of Chemical Engineers
American Institute of Electrical Engineers
American Nuclear Society
American Public Health Association
American Society of Civil Engineers
American Society of Mechanical Engineers
American Society of Safety Engineers
American Society for Testing Materials
Association of Casualty and Surety Companies
Atomic Industrial Forum, Inc.
Bureau of Explosives, AAR
Conference of State and Provincial Health Authorities of North America
Electric Light and Power Group
Health Physics Society
Institute of Radio Engineers
International Association of Governmental Labor Officials
Manufacturers Standardization Society of the Valve and Fittings Industry
Manufacturing Chemists Association
National Association of Mutual Casualty Companies
National Bureau of Standards
National Electrical Manufacturers Association
National Safety Council
Radio-Electronics-Television Manufacturers Assoc.
Scientific Apparatus Makers Association
Society of Automotive Engineers
U.S. Atomic Energy Commission
U.S. Department of Defense
U.S. Department of Health, Education and Welfare (Public Health Service)
U.S. Department of Labor (Bureau of Labor Standards)

ASA Nuclear Standards Board projects are:

N1—Glossary of Terms in Nuclear Science and Technology
N2—General and Administrative Standards for Nuclear Energy
N3—Nuclear Instrumentation Standards

3-34 EXPOSURE STANDARDS AND PROTECTION REGULATIONS

N4—Electrical Requirements for Reactors and Nuclear Power Systems and Generation and Application of Nuclear Radiation

N5—Chemical Engineering for the Nuclear Field

N6—Reactor Safety Standards

N7—Radiation Protection Standards

The American Conference of Governmental Industrial Hygienists

The **American Conference of Governmental Industrial Hygienists (ACGIH)** was organized in 1938 by a group of governmental industrial hygienists. The Conference serves as a medium for the exchange of ideas and experiences and the promotion of standards and techniques in industrial health. Its membership is limited to professional personnel in governmental agencies engaged in occupational health activities, including foreign countries.

The officers of the ACGIH include a chairman, chairman-elect, secretary-treasurer, and the executive committee. The secretary-treasurer shall be a representative of the U.S. Public Health Service and shall act as secretary of the executive committee. Its constitution states that the objectives of the ACGIH are to promote industrial hygiene in all its aspects and phases, coordinate industrial-hygiene activities by official agencies, encourage the interchange of experience among personnel in official agencies, collect and distribute information and useful data, and hold annual meetings as may be necessary to effectuate the organization's purposes.

The **ACGIH Committee on Threshold Limits** issues a list of recommendations as to the maximum concentrations of dusts, gases, vapors, mists, fumes, and radioactive materials to which workers should be exposed. The list is under constant study, and a revised set of limits is presented each year at the annual meeting.

In 1955, the Committee on Threshold Limits established values for radioactivity as: "For permissible concentrations of radioisotopes in air see 'Maximum Permissible Amounts of Radioisotopes in the Human Body and Maximum Permissible Concentrations in Air and Water,' Handbook 52, U.S. Department of Commerce, National Bureau of Standards, March 1953. In addition, see 'Permissible Dose from External Sources of Ionizing Radiation,' Handbook 59, Department of Commerce, National Bureau of Standards, September 24, 1954."

The ACGIH Committee on Industrial Hygiene Codes and Regulations has developed **A Guide for Uniform Industrial Hygiene Codes or Regulations** and supplements dealing with fluoroscopic shoe-fitting devices, dry-cleaning operations, and radioactive static eliminators. The supplement on radioactive static eliminators was reprinted and distributed throughout the printing industry by the Research Engineering Council of the Graphic Arts Industry, Inc.

Issued July, 1951, the **X-ray Shoe Fitting Guide** was designed to minimize the amount of radiation to which persons are exposed during the use of fluoroscopic shoe-fitting devices. The Guide established exposure limits for the feet, set design requirements on the construction of the unit, restricted exposures of store personnel and customers, and required the posting of a warning sign. The shoe fluoroscope, in general, has been banned in many states, although many units still in operation were designed with the ACGIH Guide as the standard.

Issued January, 1953, the **Radioactive Static Eliminator Guide** was prepared primarily to provide safeguards where static eliminators are used in fixed installations. Attention is called to portable devices for industrial and home use.

The guide requires each static eliminator to be completely labeled, to be manufactured and used in a way which will minimize the escape of radioactive materials, to be used in a manner that will keep the external dose to less than 300 mrep/week, and to be disposed of under authority supervision.

U.S. Atomic Energy Commission

Under the **Atomic Energy Act of 1954**, the **U.S. Atomic Energy Commission** is authorized to establish a licensing procedure for the control of materials and facilities essential to the atomic-energy industry, implemented by the rules, regulations, or orders and including inspections of industrial premises.

The two basic reasons for the Atomic Energy Commission's being in the regulatory field are to protect **public health and safety** and to provide for **national defense and security**. The Act prohibits the private possession or use of materials and facilities essential to the industry without a license from the Commission, requires licensees to observe such regulations and orders limiting their activities as may be issued by the Commission, and imposes controls on the dissemination of restricted data. The materials for which licenses are required are designated in the Act as "special nuclear material," "source material," and "by-product material." The Act establishes two categories of controlled facilities, "production facilities" and "utilization facilities," but leaves their exact definition to determination by the Commission. The manufacture, possession, or transfer of either type of facility without a license from the Commission is illegal. These categories have been defined recently as including nuclear reactors, facilities for the processing of special nuclear material, and facilities for separating isotopes of uranium or plutonium.

A member of the General Counsel's office of the AEC expressed the basic requirements of **AEC license regulations** as follows (Robert Lowenstein, Apr. 16, 1956, Oklahoma City, Okla.):

Basically, all the materials and the facilities regulations require that:

1. Each licensee or his staff must have suitable training or experience to possess and use the material or facility safely for the purpose for which it is requested;
2. His equipment and facilities must be adequate to protect health and minimize danger to life and property;
3. The location of the proposed use must be suitable for the purpose;
4. He may use the material or facility only for a purpose in his license;
5. He may not transfer the material or facility except to persons authorized to receive it.

The **Regulations** of the **U.S. Atomic Energy Commission** are included as parts of Title 10, Code of Federal Regulations, and are published as part of the *Federal Register*. The various parts, current and proposed, relating to health and safety, are as follows:

Part 2—Rules of Practice

Part 7—Advisory Boards

Part 8—Interpretations—Opinions of the General Counsel

Part 20—Standards for Protection against Radiation

Part 25—Access to Restricted Data

Part 30—Radioisotope Distribution; changed to Licensing of By-product Material

Part 40—Control of Source Material

Part 50—Licensing of Production and Utilization Facilities

Part 55—Operator's Licenses

Part 70—Special Nuclear Material Regulation

Of the several parts listed above, those associated with the manner of obtaining licenses contain references to the term **health and safety** in various ways in many places. Although Part 20 is the main regulation which attempts to list the essential standards for application to obtain protection against radiation, the other licensing regulations carry the concern for protecting the public health and safety into their provisions. The following is a description of some of these parts and examples of the continuous stress upon radiation protection.

Part 2, Title 10 CFR—**Rules of Practice** govern the conduct of proceedings before the Commission involving licensing and licenses (including amendments, revocations, suspensions, transfers, and renewals of licenses), intervention in such proceedings by interested persons, and how and when hearings may be obtained. The regulations propose in part that “where the public health, interest or safety requires” notice of intent by the AEC to institute proceedings to revoke a license may be omitted and the revocation may be made temporarily effective. The regulation also states, “In cases found by the Commission to be of extreme importance to the national defense and security or to the health and safety of the public, the Commission may without prior notice or hearing recapture any special nuclear material held by the licensee or enter upon and operate the licensed facility, provided that as promptly as possible and not later than 10 days from the recapture or entry, the AEC will serve . . .” a show-cause order as to why the license should not be revoked or will initiate steps to restore the material or facility.

On Feb. 28, 1957, the AEC promulgated a regulation, Part 20, Title 10 CFR—**Standards for Protection against Radiation**, establishing standards for the protection of personnel and the public against radiation hazards. The regulation applies to all persons who receive, possess, use, or transfer source material, special nuclear material, or by-product material under a general or specific license.

The regulation establishes maximum permissible limits on exposures to external radiation and concentrations of radioactive material, and maximum permissible levels of radiation in areas access to which is not controlled by the licensee. It also establishes controls over the release of radioactive material from licensed facilities. The standards incorporated in the proposed regulation are based on recommendations made by the NCRP. On the basis of present scientific knowledge and experience the permissible levels of exposure and concentration and other requirements will provide a substantial margin of safety for personnel and the public.

Other provisions of the regulation include requirements regarding personnel monitoring, caution signs and signals, waste disposal, storage of licensed material, instruction of personnel on safe procedures for handling and using licensed material, and records and reports.

The AEC issued the first draft of regulations with the statement, "Together with the proposed regulations on facilities licenses, special nuclear material, operator's licenses, and the existing source and by-product material regulations which will be revised soon, the regulations on standards for protection against radiation provide a comprehensive system for the control of radiation hazards."

Part 30, Title 10 CFR—**Licensing of By-product Material**, establishes procedures and criteria for the issuance of licenses for the use of radioisotopes. The regulation covers two types of licenses, general and specific. The general licenses "are effective without the filing of applications with the Commission or the issuance of licensing documents to particular persons." Specific licenses are issued to named persons "upon applications filed pursuant to the regulations in this part."

An application for a **specific license** to use by-product material will be approved if the applicant's proposed equipment and facilities are "adequate to protect health and minimize danger to life or property" and the applicant is qualified to use the material in such manner "as to protect health and minimize danger to life and property..." The Commission may incorporate in any license at any time by rule, regulation, or order additional requirements or conditions in order to promote the common defense and security, protect health or minimize danger to life and property, or protect restricted data or to require records and provide for inspection of activities under the license. The regulation contains sections on records, reports and inspections, modification and revocation of licenses, and enforcement.

The regulation states that "The Commission may withhold, recall or order the **withholding or recall of by-product material** from any licensee who is not equipped to observe or fails to observe such safety standards to protect health as may be established by the Commission, or who uses such materials in violation of law or regulation of the Commission, or in a manner other than as disclosed in the application therefore or approved by the Commission."

Certain items are considered as **generally licensed by-products** rather than requiring a specific license on the part of each user. A general license effectively exempts the user from most of the provisions of this part, but not all. These devices are **static-elimination units** which contain by-product material consisting of not more than 500 mc of Polonium-210 or spark-gap and **electronic tubes** containing by-product material consisting of not more than 5 mc of Cesium-137 or Nickel-63 per tube, or not more than 1 mc of Cobalt-60 per tube. A list of quantities of various by-product material as a sealed source or not as a sealed source is also included as schedule B (see Table 3-4).

Part 55, Title 10 CFR—**Operator's Licenses**, establishes procedures and minimum criteria for the issuance of licenses to operators of production and utilization facilities licenses pursuant to the Atomic Energy Act of 1954 and applies to "any individual who manipulates the controls" of any licensed facility.

The regulations in Part 55 define **controls** as "those controls ... which by manipulation or failure to manipulate singly or in combinations, could result in the release of atomic energy or radioactive material in amounts determined by the Commission to be sufficient to cause danger to the health and safety of the public."

Table 3-4. Schedule B: Quantities of By-product Materials Exempt from Specific Licensing (Generally Licensed) by the AEC

By-product material	Column I Not as a sealed source (micro- curies)	Column II As a sealed source (micro- curies)	By-product material	Column I Not as a sealed source (micro- curies)	Column II As a sealed source (micro- curies)
Antimony-124 (Sb-124).....	1	10	Palladium-103-Rhodium-103 (PdRh-103).....	50	50
Arsenic-76 (As-76).....	10	10	Phosphorus-32 (P-32).....	10	10
Arsenic-77 (As-77).....	10	10	Polonium-210 (Po-210).....	0.1	1
Barium-140-Lanthanum-140 (BaLa-140).....	1	10	Potassium-42 (K-42).....	10	10
Beryllium-7 (Be-7).....	50	50	Praseodymium-143 (Pr-143).....	10	10
Cadmium-109-Silver-109 (CdAg-109).....	10	10	Promethium-147 (Pm-147).....	10	10
Calcium-45 (Ca-45).....	10	10	Rhenium-186 (Re-186).....	10	10
Carbon-14 (C-14).....	50	50	Rhodium-105 (Rh-105).....	10	10
Cerium-144-Praseodymium- 144 (CePr-144).....	1	10	Rubidium-86 (Rb-86).....	10	10
Cesium-137-Barium-137 (CeBa-137).....	1	10	Ruthenium-106-Rhodium- 106 (RuRh-106).....	1	10
Chlorine-36 (Cl-36).....	1	10	Samarium-153 (Sm-153).....	10	10
Chromium-51 (Cr-51).....	50	50	Scandium-46 (Sc-46).....	1	10
Cobalt-60 (Co-60).....	1	10	Silver-105 (Ag-105).....	1	10
Copper-64 (Cu-64).....	50	50	Silver-111 (Ag-111).....	10	10
Europium-154 (Eu-154).....	1	10	Sodium-22 (Na-22).....	10	10
Fluorine-18.....	50	50	Sodium-24 (Na-24).....	10	10
Gallium-72 (Ga-72).....	10	10	Strontium-89 (Sr-89).....	1	10
Germanium-71 (Ge-71).....	50	50	Strontium-90-Yttrium-90 (SrY-90).....	0.1	1
Gold-198 (Au-198).....	10	10	Sulfur-35 (S-35).....	50	50
Gold-199 (Au-199).....	10	10	Tantalum-182 (Ta-182).....	10	10
Hydrogen-3 (Tritium) (H-3).....	250	250	Technetium-96 (Tc-96).....	1	10
Indium-114 (In-114).....	1	10	Technetium-99 (Tc-99).....	1	10
Iodine-131 (I-131).....	10	10	Tellurium-127 (Te-127).....	10	10
Iridium-192 (Ir-192).....	10	10	Tellurium-129 (Te-129).....	1	10
Iron-55 (Fe-55).....	50	50	Thallium-204 (Tl-204).....	50	50
Iron-59 (Fe-59).....	1	10	Tin-113 (Sn-113).....	10	10
Lanthanum-140 (La-140).....	10	10	Tungsten-185 (W-185).....	10	10
Manganese-52 (Mn-52).....	1	10	Vanadium-48 (V-48).....	1	10
Manganese-56 (Mn-56).....	50	50	Yttrium-90 (Y-90).....	1	10
Molybdenum-99 (Mo-99).....	10	10	Yttrium-91 (Y-91).....	1	10
Nickel-59 (Ni-59).....	1	10	Zinc-65 (Zn-65).....	10	10
Nickel-63 (Ni-63).....	1	10			
Niobium-95 (Nb-95).....	10	10	Beta- and/or gamma-emitting by-product material not listed above.....	1	10
Palladium-109 (Pd-109).....	10	10			

Included in the information an applicant for an AEC operator's license must submit are his education and experience, serial numbers of previous licenses, evidence that he has learned to operate the particular controls in a competent and safe manner, and a report of a medical examination, in a prescribed form indicated in Part 55.

The physical examination is required to show that the "physical condition and the general health of the applicant are not such as to be expected to cause operational errors which might endanger public health and safety."

Each operator's license shall expire 2 years after the date of its issuance.

Regulations by States

The protection of the public and the work force by state and local agencies has been achieved for the most part through the utilization of existing inspection, enforcement, and administrative rule making authority by agencies currently responsible for the protection of health from all causes.

In **California, Connecticut, Massachusetts, Michigan, Minnesota, New York, Pennsylvania, and Texas**, regulations specifically designed to effect radiation protection have been promulgated.

In addition to the eight states having radiation codes, nine states, **Alaska, Colorado, Delaware, Illinois, Kansas, New Jersey, North Dakota, South Dakota, and Wyoming**, require registration of radiation sources.

Several possible "model" codes or sets of regulations exist for a state or local agency to consider in preparing a draft of proposed regulations for effecting radiation protection within its area of jurisdiction. (State Activities in Atomic Energy, Atomic Industrial Forum, Inc., New York, N.Y., July, 1958.)

These include the following, mentioned elsewhere in more detail:

1. Appendixes A and B of **NBS Handbook 61, Regulation of Radiation Exposure**
2. **AEC Regulations—Part 20** and possibly sections from Parts 2, 8, 30, 40, 50, 55, and 70
3. Guides prepared by the **American Conference of Governmental Industrial Hygienists**
4. **American Standards** prepared by the American Standards Association
5. Other states' regulations in the same field

The following résumés are indicative of the areas covered in state codes:

Group 6, Article 53, of the General Industry Safety Orders, State of **California** Department of Industrial Relations, sets up "minimum standards for the protection of employees exposed to potentially injurious levels of ionizing radiation or potentially injurious quantities of radioactive materials" which apply to all employments and places of employment in California as defined by Labor Code Sections 6302 and 6303.

Article 53 contains the following sections: purpose, definitions, supervision and instruction, maximum permissible exposures, monitoring, maintenance of protective devices, handling of radioactive materials, storage of radioactive materials, warning signs or signals, totally protective installations, special orders for radium-dial painting, medical supervision, devices containing radioactive materials that are used as hand tools or worn on the person, and radioactive luminous compounds.

Regarding **supervision and instruction**, the California code specifies: "(a) all operations involving exposure to potentially injurious levels of ionizing radiation or potentially injurious quantities of radioactive materials shall be under the supervision of competent technical personnel . . . capable of evaluating the radiation exposure to employees and specifying such protection as required by these orders. (b) Every employee who may be regularly or frequently exposed . . . shall be instructed in the hazards . . . and methods of protecting himself and others against them."

The section on maximum permissible exposures requires **reporting of overexposures**, as follows: "measured or presumed exposure of any employee of more than

3-40 EXPOSURE STANDARDS AND PROTECTION REGULATIONS

0.3 r in any week shall be reported to the medical supervisor." An employee may be removed from further exposure when the medical supervisor finds that continued exposure is likely to injure the employee's health.

In October, 1957, the **Connecticut State Department of Health** adopted Regulation 287 of the Sanitary Code "Radiation Sources and Radioactive Materials."

The regulation applies "to the manufacture, use, storage, handling, transportation and disposal of all sources of ionizing radiation and all radioactive materials except as specifically exempted" in the regulation.

The regulation contains sections pertaining to registration, exemptions, maximum permissible doses, personnel monitoring and area surveys, records and reports, medical service, control of exposures, storage of radioactive materials, contamination control, labeling, and disposal of radioactive wastes.

Effective Dec. 1, 1957, the **Massachusetts Department of Labor and Industries** adopted *Industrial Bulletin 5*, Rules and Regulations for the Protection of the Health and Safety of Employees from Occupational Diseases Caused by Ionizing Radiation.

The rules apply "to and in all industrial establishments" which engaged in the "use, handling, processing, application, transfer, storage, and removal of all sources, materials, instruments, machines, devices, and equipment which produce, generate or emit ionizing radiation."

The rules contain sections on registration, employee qualifications, maximum permissible exposure, surveys, records, warning signs and labels, storage, medical examinations, general requirements, and exemptions.

Effective Sept. 1, 1955, the **New York State Health Department** enacted as part of the State Sanitary Code, Chapter XVI, Ionizing Radiation.

The radiation code states "every operator of a radiation installation . . . shall register such installation with the health officer having jurisdiction, prior to March 1, 1956 and before a new installation is placed in operation . . . on a form prescribed by the state commissioner of health."

The radiation code defines **installation** as "a location or facility where radiation equipment is used or where radioactive material is produced, transported, stored or used for any purpose [and] shall refer only to those installations located in a hospital; institution; medical clinic; medical office; dental clinic; dental office; veterinarian clinic; veterinary office; podiatry office; educational institution; commercial, private or research laboratory performing diagnostic procedures; shoe store; trucking, storage, messenger or delivery service establishment; or any industrial or commercial establishment not subject to supervision by the New York State Department of Labor in accordance with the laws of New York State." "Radiation installation" shall include whether or not it is specifically stated, "any facility where radiation is applied intentionally to a human."

In addition to the registration requirement, the radiation code includes sections on maximum permissible doses, supervision, personnel protection, medical examinations, radiation instruments, handling of cadavers containing radioisotopes, monitoring of radiation installations, therapy rooms, warning signs, accounting for radioactive materials, radiation illnesses, injuries, emergencies and accidents, electrical hazards, and vacated premises.

In 1955, an Act was passed amending the **New York State Public Health Law** by

adding to Section 201 ("Functions, Powers and Duties of the Department") a new paragraph: "(s) supervise and regulate the public health aspects of the use of ionizing radiation and the handling and disposal of radioactive wastes."

Effective Dec. 15, 1955, the Board of Standards and Appeals of the **New York State Department of Labor** enacted Industrial Code Rule 38, Radiation Protection.

Code Rule 38 requires that "the owner of every installation shall register the same or cause it to be registered with the Industrial Commissioner. Such **registration** shall be in a form acceptable to the Commissioner." A registration form has been prepared by the Department.

Under the section on Application, the rule "applies to every place and every operation where any employee in the course of his work may be exposed to radiation in excess of one-tenth the permissible weekly dose as set forth in this rule." Specifically exempted from this section are places and installations subject to the provisions of the New York State sanitary code.

Operations involving only the use of specified devices, appliances, and materials are exempted from the application of the Code Rule.

Code Rule 38 contains sections on a finding of fact that radiation use in industry involves elements of danger, application of the rule, exemptions from the application, definitions, registration requirements for installations and mobile sources, control of exposure including requiring the services of a **radiation safety supervisor**, dose limits and variations, survey requirements, record-keeping requirements, personnel monitoring, caution labels and signs, storage of radioactive material, and a statement on severability.

Article 6 of the **New York City** Sanitary Code became effective June 15, 1958. New York City is not subject to the jurisdiction of the New York State Department of Health and has its own Health Department. Article 6 includes by reference all of the applicable NBS Handbooks and the various Federal transportation regulations. Every owner or operator of a "Radiation Installation" must register and pay a \$15 registration fee (renewable every two years at \$10).

On Oct. 20, 1956, Regulation 433, Radiation Protection, was adopted by the Advisory Health Board of **Pennsylvania**.

This regulation is patterned after Appendix B of NBS Handbook 61 but is not identical with it.

As in NBS Handbook 61, the Pennsylvania regulation required **registration** of radiation machines and radioactive material. The person "owning" or using such sources shall notify the Department of the fact in writing, "within 30 days following the commencement thereof, and every two years thereafter."

The section on radiation information **labeling** was modeled after the same section in the first draft (July, 1955) of AEC's proposed Part 20, Title 10, CFR. Certain modifications were included to relate to other sections of the Pennsylvania regulations.

Regulation 433 contains sections on scope, application, definitions, exemptions, standards, registration, maximum permissible dose, personnel monitoring, area radiation surveys, radiation-exposure records and reports, responsibility, storage of radioactive materials, radioactive-contamination control, radiation-information labeling, disposal of radioactive wastes, and technical standards and general information to be used in achieving the requirements of the regulations.

3-42 EXPOSURE STANDARDS AND PROTECTION REGULATIONS

The **Texas** State Board of Health adopted Regulations on Radiation Exposure, effective Sept. 1, 1956.

The regulation is effectively identical with Appendix B of NBS Handbook 61. Any differences are minor and attributable to the administrative pattern of the state government.

FEDERAL REGULATIONS APPLYING TO THE TRANSPORTATION OF RADIOACTIVE MATERIALS

(Excerpted from a handbook of the same title prepared by the AEC Division of Construction and Supply, Traffic Management Section, July, 1955, with modifications based on recent changes in the regulations.)

This section is designed to be of assistance in locating and understanding Federal regulations applying to the shipment of radioactive materials. Portions of pertinent regulations promulgated by the Interstate Commerce Commission, Civil Aeronautics Board, U.S. Coast Guard, and Post Office authorities are reproduced together with comments believed to be helpful in determining the proper application of the regulations. An explanation of technical terms not defined in the regulations is also given. At the end, auxiliary information to shippers includes a number of miscellaneous items that should be helpful in determining the manner of shipment and shipping responsibilities.

The following limitations should be noted:

1. Both the selection of excerpts and the comments included in this section are unofficial as far as the regulating agencies are concerned.
2. Comments given herein, while based on the best information available to the authors of this section, are not based on court decisions nor on official interpretations by the regulating agencies.
3. The regulations excerpted herein are subject to change with due notice in the *Federal Register* or, in the case of U.S. postal regulations, in the *Postal Bulletin*.

Persons responsible for the conformance of shipments of radioactive materials to Federal transportation regulations should have access to and be familiar with the complete regulations applicable to such shipments.

Detailed information on many of the basic problems involved in the transportation of radioactive materials is contained in a preliminary report, "Physical, Biological, and Administrative Problems Associated with the Transportation of Radioactive Substances," by Robley D. Evans, Chairman of the Subcommittee on Shipment of Radioactive Substances, Committee on Nuclear Science, Division of Physical Sciences, National Research Council. This report, issued as Publication 205, is available for \$1 at the Publications Office of the National Research Council, Washington 25, D.C.

Transportation of radioactive materials in interstate commerce by land or water is subject to regulations of the Interstate Commerce Commission. The ICC does not issue separately its regulations applying specifically to shipments of radioactive materials (or to any other commodity or group of commodities). The complete regulations covering the packaging, labeling, and transportation of dangerous articles are published as Title 49, Parts 71 to 78 of the Code of Federal Regulations. Between revisions, annual pocket supplements are issued. Amendments subsequent to the period covered by the most recent revision or annual supplement may be obtained from the daily issues of the *Federal Register*. All of these are for sale by the Superintendent of Documents, U.S. Government Printing Office, Washington 25, D.C.

The ICC regulations are published also by the Bureau of Explosives of the Association of American Railroads, H. A. Campbell, Agent, 30 Vesey Street, New York 7, N.Y.,

as "Tariff No. 10, Publishing Interstate Commerce Commission Regulations for Transportation of Explosives and Other Dangerous Articles . . .," and by the Tariff Bureau of the American Trucking Associations, Inc., F. G. Freund, Agent, 1424 16th St., N.W., Washington 6, D.C., as "Motor Carriers' Explosives and Dangerous Articles Tariff No. 8."

Transportation of radioactive materials in aircraft is regulated by the Civil Air Regulations, "Transportation of Explosives and Other Dangerous Articles" is also published separately and is for sale by the Superintendent of Documents, U.S. Government Printing Office.

In addition to the minimum restrictions on transportation by air prescribed by the Civil Aeronautics Board, various individual airlines further restrict the conditions under which they will accept radioactive and other materials for transportation. A compilation of these restrictions is issued as "Official Air Transport Restricted Articles Tariff," by Emery F. Johnson, Agent, National Airport, Washington 1, D.C.

Transportation of radioactive materials by water is subject to regulations prescribed by the Commandant of the U.S. Coast Guard. Current regulations applicable to radioactive materials appear in Title 46, Part 146 of the Code of Federal Regulations, as amended, see *Federal Register*, July 17, 1952, page 6460 ff., and December 31, 1952. The U.S. Coast Guard regulations covering transportation, storage or stowage of dangerous articles on ships are also published by the Bureau of Explosives, H. A. Campbell, Agent, 30 Vesey Street, New York 7, N.Y., as "Water Carrier Tariff No. 6."

Regulations governing the transportation of radioactive materials in the U.S. mails are given in the *U.S. Postal Guide*. In the 1951 edition, they appear on page 51 of Part I.

Persons responsible for export shipments of radioactive materials should determine what regulations may apply to such shipments in countries to which they are made. See paragraphs 73.8 and 73.9 of the ICC regulations, excerpted below, for statements relative to Canadian and other foreign shipments.

The regulations issued by the Civil Aeronautics Board and by the U.S. Coast Guard embody the packaging and labeling requirements of the Interstate Commerce Commission. U.S. Postal Regulations permit transportation by mail of only those radioactive materials which under ICC regulations are exempt from specification packaging and labeling. (Such packages must, however, be labeled as prescribed by the Postal Regulations when submitted for transportation in U.S. mail.) For these reasons, definitions, regulations, etc., common to the ICC and another regulatory agency are discussed only where they first appear in the ICC regulations.

ICC REGULATIONS GOVERNING THE TRANSPORTATION OF RADIOACTIVE MATERIALS

The transportation of radioactive materials moving in interstate commerce by rail, water or by public highway (except in U.S. mail) is regulated by the Interstate Commerce Commission. Some states extend the ICC regulations to intrastate transportation. In addition, local authorities may impose additional limitations upon the transportation of radioactive materials, as in the case of their movement through tunnels or within port areas.

The **Interstate Commerce Commission Regulations** covering the Transportation of Explosives and Other Dangerous Articles include eight parts of Title 49 of the Code of Federal Regulations, as follows:

Part 71.—General Information and Regulations.

Part 72.—Commodity List of Explosives and Other Dangerous Articles . . .

3-44 EXPOSURE STANDARDS AND PROTECTION REGULATIONS

Part 73.—Regulations Applying to Shippers.

Part 74.—Regulations Applying Particularly to Carriers by Rail Freight.

Part 75.—Regulations Applying to Carriers by Rail Express . . .

Part 76.—Regulations Applying to Rail Carriers in Baggage Service . . .

Part 77.—Regulations Applying to Shipments Made by Way of Common, Contract or Private Carriers by Public Highway.

Part 78.—Shipping Container Specifications.

In addition to a limited number of excerpts from Parts 71-72 and Parts 74-77, comments on some of the provisions of these parts are also given. Specifications for shipping containers, given in Part 78, prescribed by these regulations, are not excerpted.

This section is concerned primarily with regulations directly applicable to the shipper of radioactive materials. Regulations of particular interest, including all of the regulations in Part 73 applying specifically to radioactive materials, are excerpted below. Some excerpts are followed by comments intended to facilitate the interpretation or the application of the regulations to which the comments refer.

In this section, excerpts from the regulations of the Interstate Commerce Commission and of other Federal agencies are distinguished from comments on these regulations by subparagraphs and indentation. Omissions from a paragraph are represented by “. . .”, and omission of one or more paragraphs of a section is represented by “* * * * *”. Omission of entire sections is not, in general, indicated.

ICC REGULATIONS

PART 71.—GENERAL INFORMATION AND REGULATIONS

71.1. *Plan of the regulations in Parts 71-78:*

(a) Regulations in Parts 71-78 cover preparation of explosives and other dangerous articles for transportation by common carriers by rail freight, rail express, rail baggage, highway or water, construction of containers, packing, weight, marking, labeling when required, billing and shipper's certificate of compliance with these regulations; also cars, loading, storage, billing, placarding, and movement thereof by carriers by rail.

(b) Regulations for equipment and operation of motor vehicles on the highways are published in separate issues of the Commission.

71.2. *Act of Congress:*

(a) Section 834, Title 18 of the United States Code, approved June 25, 1948 (Pub. Law 772, 80th Cong.), provides that whoever knowingly delivers to any common carrier engaged in interstate or foreign commerce by land or water, or carries upon any car or vehicle operated by any common carrier engaged in interstate or foreign commerce by land, any explosive or other dangerous article specified in section 832, under any false or deceptive marking, description, invoice, shipping order or declaration, or without informing the agent of such carrier, in writing, of the true character thereof, or does not plainly mark on the outside of every package containing explosives or other dangerous articles the contents thereof, shall be fined or imprisoned, or both, as provided in this act.

71.3. *Changes in the regulations, shippers by rail, highway, and water, and carriers by rail and highway:*

(a) Section 835 of the act of June 25, 1948, authorizes the Commission to formulate regulations for the safe transportation of explosives and other dangerous articles

and either upon its own motion, or upon application by any interested party, to make changes or modifications in such regulations, made desirable by new information or altered conditions. It further provides that in the execution of sections 831-835 of the Act the Commission may utilize the services of the Bureau for the Safe Transportation of Explosives and Other Dangerous Articles (hereinafter called Bureau of Explosives). The Bureau of Explosives will make inspections and conduct investigations and will confer with manufacturers and shippers with a view to determining what regulations will within reasonable limits afford the highest degree of safety in preparing and packing explosives and other dangerous articles for transportation by carriers by rail, highway, or water. The Commission will give due weight to the expert opinions thus obtained. Reports of these investigations will be made to the Commission with recommendations.

(b) Specifications for shipping containers, methods of packing for shipment, and other regulations will be considered and prescribed from time to time by orders effective as conditions may appear to warrant.

71.6. *Approved changes; notice:*

(a) The act of June 25, 1948, requires that notice of 90 days after formulation and publication should be given of the effective date of new or modified regulations, unless a shorter time is authorized by the Commission. The authority to establish amended regulations upon less than 90 days' notice will be exercised only in instances where special and peculiar circumstances or conditions fully justify it.

71.7. *Public hearings:*

(a) Public hearings concerning regulations contained in Parts 71-78 will be held by the Commission at sufficiently frequent intervals. At these hearings evidence may be introduced in favor of proposed changes or additions and protest against the adoption thereof will also be heard. Final action also may be taken by the Commission without hearing, following 20 days' notice by the Commission of proposed changes or additions, or without such notice, as conditions appear to warrant.

71.11. *Transportation by carriers by water:*

(a) When the transportation of a shipment involves movement by carrier by water, the applicable provisions of Parts 71-78 must be observed by the shipper.

71.12. *Export shipments by domestic carriers by rail and motor vehicles:*

(a) Explosives and other dangerous articles authorized to be exported from the United States when packed, marked, labeled, and described, in accordance with rules and regulations in force at destination ports, must not be offered to any common carrier by rail or motor vehicle for domestic transportation unless in full accordance with the regulations in Parts 71-78.

(b) Except for the requirements of 77.817 and 77.823, the provisions of Parts 71-78 do not apply to such transportation by motor vehicle or water as may be necessary to effect transfer of export shipments from place of shipment to other places within the same port area or delivery to a water carrier within the same port area (including contiguous harbors). Further transportation of such export shipments by connecting water carrier shall be subject to the regulations prescribed by the Commandant of the Coast Guard.

3-46 EXPOSURE STANDARDS AND PROTECTION REGULATIONS

PART 72.—COMMODITY LIST OF EXPLOSIVES AND OTHER DANGEROUS ARTICLES CONTAINING THE SHIPPING NAME OR DESCRIPTION OF ALL ARTICLES SUBJECT TO PARTS 71-78

72.1. *Proper shipping name:*

(a) The proper shipping name which must be used and shown on outside shipping containers appears [in the commodity list, 72.5] in roman type (not italics). . . .

The words between brackets above do not appear in the ICC regulations.

72.3. *Labels required and prohibited articles:*

(a) Section 72.5 of this part also shows the kind of label when required on shipments of explosives and other dangerous articles and the articles which are prohibited for transportation.

Section 72.5, referred to in 72.3, is a list of explosives and other dangerous articles to which the regulations of Parts 71-78 apply. Items are listed in alphabetical order and for each item there is given the proper shipping name, the class of hazard, cross references to sections specifying exemptions and packing, color of label required if not exempt, and maximum quantity in one outside container for shipment by rail express. All radioactive materials are classed as poison, class D, and are properly shipped as "radioactive materials." Blue or red label is required, as specified in section 73.414.

PART 73.—REGULATIONS APPLYING TO SHIPPERS

73.1. *Purpose of the regulations in Parts 71-78:*

(a) To promote the uniform enforcement of law and to minimize the dangers to life and property incident to the transportation of explosives and other dangerous articles by common carriers engaged in interstate or foreign commerce, the regulations in Parts 71-78 are prescribed to define these articles for transportation purposes, to state the precautions that must be observed by the shipper in preparing them for shipment by rail freight, rail express, rail baggage, highway, or by carrier by water. It is the duty of each such shipper to make the prescribed regulations effective and to thoroughly instruct employees in relation thereto.

Except as otherwise specified, each of the regulations given in Part 73 is applicable to articles prepared for shipment by any of the modes of transportation named in paragraph 73.1 (a).

(b) Explosives and other dangerous articles may be offered to carriers for transportation provided the articles are in proper condition for transportation, are as defined, and are packed, marked, labeled, described, certified, and otherwise as provided for in Parts 71-78 for acceptable articles for transportation by rail freight, rail express, rail baggage, highway, or water. Articles must be loaded and stayed according to regulations in Parts 71-78 applying to carriers by rail. Methods of manufacture, packing, and storage, insofar as they affect safety in transportation, must be open to inspection by a duly authorized representative of the initial carrier or of the Bureau of Explosives. Shipments that do not comply with the regulations in Parts 71-78 must not be offered for transportation.

73.2. *Classification; dangerous articles:*

(a) Dangerous articles other than explosives having more than one hazardous characteristic, as defined by the regulations in Parts 71-78 must be classified according

to the greatest hazard present, except those articles which are also poisons, class A, or class D, which must be classified according to both dangerous characteristics as defined herein.

This applies to all radioactive materials which have also another "hazardous" characteristic, unless exempt from specification packing, marking, and labeling under the provisions of 73.392. An example is radioactive metallic sodium which is classed also as a flammable solid. If the radioactivity exceeds that specified in 73.392 (a) (2), it is subject both to the regulations applying to radioactive materials and to flammable solids. (No quantity of metallic sodium is exempt from specification packing, marking, and labeling as a flammable solid. See 72.5 and 73.206 for details.) 73.402 (a) (2) provides that such a package shall carry both the appropriate radioactive materials warning label specified in 73.414 and the yellow warning label for a flammable solid specified in 73.406. A similar situation exists in the case of radioactive sodium alloy.

Other examples of materials to which 73.2 may be applicable are powdered thorium metal (see 73.226) and tritium gas. While no quantity of powdered thorium metal is exempt from specification packing, marking, and labeling as a flammable solid, because of its low activity, a sizable quantity (the magnitude of which depends upon the interpretation of 73.392 (a) (2)) is exempt from specification labeling as a radioactive material.

73.7. *United States Government shipments:*

* * * * *

(b) Shipments of radioactive materials, made by the Atomic Energy Commission, or under its direction or supervision, which are escorted by personnel specially designated by the Atomic Energy Commission, are exempt from the regulations in Parts 71-78.

While shipments of radioactive materials made under the provisions of paragraph 73.7 (b) are legally exempt from all of the regulations in Parts 71-78, no such shipment should be made under conditions which provide less over-all protection to the carrier, to the public, or to the consignee than would be provided by the ICC regulations if the shipment were not exempt. In particular, shipment under escort should not be permitted to cover careless packaging and handling.

Shipments of radioactive materials by AEC contractors may be made under the provisions of paragraph 73.7 (b) provided they are escorted by personnel specifically approved by the Atomic Energy Commission for the shipments involved. Such approval shall be obtained through normal contractual channels from the Manager of Operations or his authorized representative.

Specific approval by the Bureau of Explosives or by the Interstate Commerce Commission of the shipment of radioactive materials under the provision of 73.7 (b) is not required.

73.8. *Canadian shipments:*

(a) Explosives and other dangerous articles, as defined in Parts 71-78 which are packed, marked, labeled, and loaded, in conformity with the regulations of the Board of Transport Commissioners for Canada, may be transported from point of entry in the United States to their destination in the United States or through the United States en route to a point in Canada. . . .

(b) Specification containers made and maintained in full compliance with corresponding specifications prescribed by the Board of Transport Commissioners for Canada in Its Regulations for the Transportation of Explosives and Other Dangerous Articles by Freight, and Specifications for Shipping Containers, and marked in ac-

3-48 EXPOSURE STANDARDS AND PROTECTION REGULATIONS

cordance therewith, CRC, etc., may be used for shipment of explosives and other dangerous articles offered for transportation by carriers by rail freight, rail express, highway, or water.

Canadian regulations for shipment of radioactive materials by rail are essentially identical with those of the U.S. Interstate Commerce Commission, including arrangement and numbering. Special permits or other special arrangements are approved by the Board of Transport Commissioners. Requests involving containers for which permits have been issued in the United States by the Bureau of Explosives will be facilitated by including with other pertinent information the Bureau of Explosives permit numbers of the containers to be used.

Shipments of radioactive materials in Canada, made by Atomic Energy of Canada Limited, or under its direction or supervision, which are escorted by personnel specially designated by Atomic Energy of Canada Limited, are exempt from the regulations of the Board of Transport Commissioners.

73.9. *Import and export shipments:*

(a) Import shipments of explosives and other dangerous articles offered in the United States in original packages for transportation by carriers by rail freight, rail express, motor vehicle, or water must comply with all requirements of the regulations in Parts 71-78. The importer must furnish with the order to the foreign shipper, and also to the forwarding agent at the port of entry, full and complete information as to the packing, marking, labeling, and other requirements, as prescribed in Parts 71-78. The forwarding agent must file with the initial carrier in the United States a properly certified shipping order or other shipping paper as prescribed in this part. Except for the requirements of 77.817 and 77.823, the provisions of Parts 71-78 do not apply to such transportation by motor vehicle or water as may be necessary to effect transfer of import shipments from place of discharge to other places within the same port area or delivery to a water carrier within the same port area (including contiguous harbors). Further transportation of such import shipments by connecting water carrier shall be subject to the regulations prescribed by the Commandant of the Coast Guard.

(b) Shipments of explosives and other dangerous articles offered for transportation by common carrier by water from the United States, its insular possessions, or dependencies, destined to such insular possessions or territory, dependencies, or to a foreign country, must be packed, marked, labeled and described in accordance with the rules and regulations in force at destination ports or as prescribed in Parts 71-78.

73.11. *Violations and accidents to be reported:*

(a) Consignees must report promptly to the Bureau of Explosives all instances of improper stowing and broken, leaking, or defective containers of explosives or other dangerous articles in shipments received by them.

(b) The Bureau of Explosives, upon receipt of reports from consignees, should promptly report to the shipper full particulars covering all such cases.

SUBPART A.—PREPARATION OF ARTICLES FOR TRANSPORTATION BY CARRIERS BY RAIL FREIGHT, RAIL EXPRESS, HIGHWAY, OR WATER

73.27. *Rail express limitations:*

* * * * *

(c) When several dangerous articles are placed in one outside package without violating the regulations, the combined quantity of any one group must not exceed the lowest limit prescribed for any one of the articles of that group that is included.

73.29. *Empty containers:*

(a) Empty cylinders, barrels, kegs, drums, or other containers except carboys (see paragraph (c) of this section) previously used for the shipment of any explosive or other dangerous article, as defined in this part, if authorized for reuse must have all openings including removable heads, filling and vent holes, tightly closed before being offered for transportation. Small quantities of the material with which containers were loaded may remain in "empty" containers and when the vapors remaining therein are unstable, it is permissible to add sufficient inert gas to render the vapors stable.

* * * * *

(e) All containers and accessories which have been used for shipments of radioactive materials when shipped as empty must be sufficiently free of radioactive contamination so as to conform to the conditions of paragraph (a) (1), (2), and (3) of 73.392 of this part.

(f) Containers shipped as "empty" must have the old labels prescribed by this part removed, obliterated, destroyed, or completely covered by a square white label as described in 73.413 of this part, measuring not less than six inches on each side, and bearing thereon the word "EMPTY" in letters not less than one-inch high. This does not apply to carload or truckload shipments to be unloaded by consignee. Since the use of warning labels on used containers shipped as "Empty" is prohibited, it is desirable that in cases in which the consignee may be inconvenienced by unsuspected radioactive content of such containers he should be directly informed of the shipment. Such shipments may also be entered on bills of lading or other shipping papers as "Empty containers which have contained radioactive materials." However, such description does not provide positive assurance that the information will appear on papers reaching the consignee, since in transcribing the description from one set of papers to another the carrier may shorten the description to the minimum required for tariff classification, "Empty Containers."

73.30. *Loading and placarding of cars by shippers and unloading of cars by consignees:*

(a) When shipments of explosives or other dangerous articles are loaded into cars by shippers, or unloaded from cars by the consignee or his duly authorized agent, the applicable provisions of Part 74 must be complied with. See 74.538 for loading and storage chart.

SUBPART D.—FLAMMABLE SOLIDS AND OXIDIZING MATERIALS; . . .

73.226. *Thorium metal, powdered:*

(a) Thorium metal, powdered, must be packed in specification containers as follows:

1. Spec. 15A or 15B. Wooden boxes with inside metal containers, tightly and securely closed by push-in covers held in place by soldering at least at four points, or in screw-cap type metal cans. Inside containers must not exceed 10 pounds net each. Gross weight of outside packages must not exceed 75 pounds each.

SUBPART G.—POISONOUS ARTICLES; DEFINITION AND PREPARATION

73.391. *Radioactive materials class D Poison; definition:*

(a) For the purpose of Parts 71-78 radioactive material is any material or combination of materials that spontaneously emits ionizing radiation. For the purpose of Parts 71-78, radioactive materials are divided into three groups according to the type of rays emitted at any time during transportation, as follows:

3-50 EXPOSURE STANDARDS AND PROTECTION REGULATIONS

1. *Group I.*—Radioactive materials that emit gamma rays only or both gamma and electrically charged corpuscular rays.
2. *Group II.*—Radioactive materials that emit neutrons and either or both the types of radiation characteristic of Group I materials.
3. *Group III.*—Radioactive materials that emit electrically charged corpuscular rays only; i.e., alpha or beta, etc., or any other that is so shielded that the gamma radiation at the surface of the package does not exceed 10 milliroentgens for 24 hours at any time during transportation.

Since all materials are radioactive to some degree, it must be presumed that the above definition of a radioactive material is intended to apply only to materials having a history which suggests that they have a higher degree of radioactivity than is normal to materials in common usage. For example, materials which may have been subjected to a considerable quantity of activating radiation, materials which may have been contaminated with radioactive materials, or materials from mines containing radioactive ores must be considered to come under this definition unless it is known that their radioactivity is within the range of that of materials not normally considered to be radioactive.

For purposes of ICC regulations, packages of radioactive materials are classified as Group I, Group II, or Group III to facilitate the statement of regulations covering labeling and handling. Packages classed as Group I or Group II require special precautions in transit and in storage to protect personnel and photographic film from radiations emitted from the packages. The stipulation "at any time during transportation" in describing a Group III package is made necessary by the fact that the gamma radiation from some packages will increase during transit due to the formation of gamma-emitting daughter products of the radioisotope being shipped.

(b) Not more than 2,000 millicuries of radium, polonium or other members of the radium family of elements, and not more than 2,700 millicuries (disintegration rate of 100,000 million (10^{11}) atoms per second) of any other radioactive substance may be packed in one outside container for shipment by rail freight, rail express, or highway, except by special arrangements and under conditions approved by the Bureau of Explosives or except as specifically provided in subparagraph (c) of this section.

NOTE 1.—For purposes of Parts 71-78 of this chapter, one millicurie is that amount of any radioactive material which disintegrates at the rate of 37 million atoms per second.

(c) Not more than 300 curies of solid cesium 137, cobalt 60 or iridium 192, may be packed in one outside container for shipment by rail freight, rail express, or highway, except by special arrangements and under conditions approved by the Bureau of Explosives.

Activities of shipments of all radioactive materials generally are measured in curies or in sub-multiples of curies. One curie is equivalent to 1,000 millicuries or to 1,000,000 microcuries. From Note 1 above, "that amount of any other radioactive substance which disintegrates at the rate of 100,000 million atoms per second" is 2.7 curies.

Paragraph 73.391 (b) is generally interpreted as permitting the shipment of 2 curies (2 grams) of Ra^{226} in a single package, regardless of the fact that, as ordinarily shipped, the package would contain approximately two curies each of a number of radioactive decay products. That this was the intent of the National Research Council subcommittee formulating the original draft of these regulations, is indicated by the following statement from page 39 of NRC Publication 205: "The maximum quantity of radium whose shipment is permitted in a single container, as in a neutron source, is 2000 mc. or 2 gm. Ra." Field comment indicates, however, that some AEC installations interpret the two curies as applying to all of the radioactive material believed to be in the package. The latter interpretation receives some support from 73.27 (b). The interpretation

of this paragraph as applied to other radioactive substances, e.g., uranium, thorium, strontium 90, zirconium 95, is also controversial.

73.392. Exemptions for radioactive materials:

(a) Radioactive materials are exempt from prescribed packaging, marking, and labeling requirements provided they fulfill all of the following conditions:

1) The package must be such that there can be no leakage of radioactive material under conditions normally incident to transportation.

2) The package must contain not more than 0.1 millicurie of radium, or polonium, or that amount of strontium 89, strontium 90, or barium 140 which disintegrates at a rate of more than 5 million atoms per second; or that amount of any other radioactive substance which disintegrates at a rate of more than 50 million atoms per second.

That amount of material which disintegrates at the rate of 5 million atoms per second is 0.135 millicurie; and that which disintegrates at the rate of 50 million atoms per second is 1.35 millicuries.

3) The package must be such that no significant alpha, beta, or neutron radiation is emitted from the exterior of the package and the gamma radiation at any surface of the package must be less than 10 milliroentgens for 24 hours.

(b) Manufactured articles other than liquids, such as instrument or clock dials or electronic tubes and apparatus, of which radioactive materials are a component part, and luminous compounds, when securely packed in strong outside containers are exempt from specification packaging, marking, and labeling requirements provided the gamma radiation at any surface of the package is less than 10 milliroentgens in 24 hours.

1) Switchboard or similar apparatus containing electronic tubes, of which radioactive materials are a component part, are exempt from specification packaging, marking, and labeling requirements when shipped in carload or truckload lots or when transported by private motor carrier provided the gamma radiation at any readily accessible surface of the units, when prepared for shipment, does not exceed 50 milliroentgens in 24 hours.

(c) Radioactive materials such as ores, residues, etc., of low activity packed in strong tight containers are exempt from specifications packaging and labeling requirements for shipment in carload lots by rail freight only provided the gamma radiation or equivalent will not exceed 10 milliroentgens per hour at a distance of 12 feet from any surface of the car and that the gamma radiation or equivalent will not exceed 10 milliroentgens per hour at a distance of 5 feet from either end surface of the car. There must be no loose radioactive material in the car, and the shipment must be braced so as to prevent leakage or shift of lading under conditions normally incident to transportation. The car must be placarded by the shipper as provided in 74.541 (b) and 74.553, except when handling is supervised by the Atomic Energy Commission. Shipments must be loaded by consignor, and unloaded by consignee.

Paragraph 73.392 (c) provided exemption from specification packaging, marking, and labeling for shipments of packaged low activity materials by rail freight, in carload lots only, subject to the conditions given therein. Such exemption is provided independently of the general conditions for exemption from specification packaging, marking, and labeling given in paragraph 73.392 (a). However, such materials are not exempt from specification packaging, labeling and marking when shipped by motor truck or by railway express unless they meet the requirements of 73.392 (a). See, for example, 77.815 (a), (b), and (c).

The term "low activity material" is not defined by the ICC. The wording of 73.392 (c) and of 74.532 (j) (1) implies that any gamma emitting material, a full carload of

3-52 EXPOSURE STANDARDS AND PROTECTION REGULATIONS

which does not produce radiation in excess of 10 mr./hr. at a distance of 12 feet from any surface of the car may be considered low activity material for this purpose. Limitation of the radiation from either end of the car to 10 mr./hr. at 5 feet may be achieved either by spacing or by the use of shielding.

The term "no loose material" is not defined by the ICC. The usage here and in paragraph 74.532 (j) implies that "loose material" means amounts of material which may be introduced as a result of spillage, either within the car or onto the containers or equipment which are brought into the car. The level of gamma radiation which may exist in the car after unloading is specified in section 73.395.

73.393. *Packing and shielding:*

(a) Radioactive materials that present special hazards due to their tendency to remain fixed in the human body for long periods of time (i.e., radium, plutonium, and radioactive strontium, etc.) must, in addition to the packaging hereinafter prescribed, be packed in inside metal containers specification 2R, or other container approved by the Bureau of Explosives.

Specifications for 2R containers are given in Section 78.34 of the ICC Regulations (not excerpted in this section). The requirement that packages containing radium, plutonium, etc., be packed in inside containers of this specification is not considered to apply to cases in which these radioisotopes occur in low activity materials as discussed in 73.392 (c) or as impurities in other radioactive materials.

(b) All radioactive materials must be so packed and shielded that the degree of fogging of undeveloped film under conditions normally incident to transportation (24 hours at 15 feet from the package) will not exceed that produced by 11.5 milliroentgens of penetrating gamma rays of radium filtered by $\frac{1}{2}$ inch of lead.

The quantity of gamma radiation referred to, 11.5 milliroentgens, is that measured after filtration by the lead. Photographic emulsions are more sensitive to gamma radiations of quantum energies less than about 0.15 Mev than to harder or more penetrating gamma radiation of higher quantum energies. Filtration of radium gamma radiation by this thickness of lead removes practically all of the quanta with energies as low as 0.15 Mev and provides an arbitrary standard of photographic effectiveness with which other gamma radiation may be compared.

For cases in which the gamma emission from the radioactive materials within the package contains a strong component of quantum energy higher than about 0.2 or 0.3 Mev, and a substantial amount of lead shielding is used in bringing the radiation from the package down to prescribed limits, the gamma radiation emitted from the package may be considered to have the same photographic effectiveness per milliroentgen of exposure as gamma radiation from radium filtered by $\frac{1}{2}$ inch of lead. Ordinarily, for Group I packages, this is the case.

The following table, adapted from Table IX, p. 48, NRC Publication 205, gives the relative sensitivity of photographic emulsions to gamma radiations of different quantum energies below 0.15 Mev as compared to sensitivity to radium radiation filtered as specified above:

Quantum energy.....	0.14	0.12	0.09	0.075	0.06	0.045	0.03
Relative sensitivity....	1.2	2.3	4.9	8.6	14	21	24

As an example of the application of these values to paragraph 73.393 (b), if a package emits gamma radiation of which the effective quantum energy is 0.09 Mev, the maximum gamma radiation permitted at 15 feet from the package is 11.5/4.9 milliroentgens per 24 hours. For a more complete discussion, the reader is referred to NRC Publication 205, pp. 15-24 and 48-49.

(c) The design and preparation of the package must be such that there will be no significant radioactive surface contamination of any part of the container.

(d) The smallest dimension of any outside shipping container for radioactive materials must be not less than 4 inches.

(e) All outside shipping containers must be of such design that the gamma radiation will not exceed 200 milliroentgens per hour or equivalent at any point of readily accessible surface. Containers must be equipped with handles and protective devices when necessary in order to satisfy this requirement.

(f) The outside shipping container for any radioactive material, unless specifically exempt by 73.392 or unless approved by the Bureau of Explosives, shall be as follows:

(1) Spec. 15A or 15B (78.168 or 78.169 of ICC Regulations). Wooden boxes. Authorized for not more than 2,700 millicuries; (2) Spec. 12B (78.205 of ICC Regulations). Fibreboard boxes. Authorized for not more than 2,700 millicuries; (3) Spec. 21A or 21B (78.222 or 78.223 of ICC Regulations). Fibre drums. Authorized for not more than 2,700 millicuries; (4) Spec. 6A, 6B, or 6C. 17C or 17H (single-trip) (78.97, 78.98, 78.99, 78.115 or 78.118 of ICC Regulations). Metal barrels or drums. Authorized for not more than 2,700 millicuries; (5) Spec. 55 (78.250 of ICC Regulations). Metal-encased, lead-shielded containers. Authorized for not more than 300 curies (see 73.391). Containers must be equipped with a seal.

Specifications for these containers are given in Part 78 of ICC regulations. These specifications are not excerpted in this section.

(g) Radioactive materials Group I, liquid, solid, or gaseous, must be packed in suitable inside containers completely surrounded by a shield of lead or other suitable material of such thickness that at any time during transportation the gamma radiation at one meter (39.3 inches) from any point on the radioactive source will not exceed 10 milliroentgens per hour. The shield must be so designed that it will not open or break under conditions incident to transportation. The minimum shielding must be sufficient to prevent the escape of any primary corpuscular radiation to the exterior of the outside shipping container.

"... at one meter from any point on the radioactive source ..." is interpreted by the AEC to mean "at one meter from the nearest point on the source."

(h) Radioactive materials Group II, liquid, solid, or gaseous, must be packed in suitable inside containers, completely shielded so that at any time during transportation the radiation measured at right angles to any point on the long axis of the shipping container will not exceed the limits specified in subparagraphs (1) to (4) of this paragraph. The shielding must be designed so as to maintain its efficiency under conditions normally incident to transportation and must provide personnel protection against fast or slow neutrons and all other ionizing radiation originating in the radioactive materials or any part of the aggregate constituting the complete package.

(1) Gamma radiation of 10 mrhm.

(2) Electrically charged corpuscular radiation which is the physical equivalent (see note 1 of this paragraph) of 10 mrhm. of gamma radiation.

(3) Neutron radiation which is the physical equivalent (see note 1 of this paragraph) of 2 mrhm. of gamma radiation.

(4) If more than one of the types of radiation named in subparagraphs (1), (2), and/or (3) of this paragraph is present the radiation of each type must be reduced by shielding so that the total does not exceed the equivalent of subparagraphs (1), (2), or (3) of this paragraph.

NOTE 1.—For purposes of Parts 71-78 the "physical equivalent" of a roentgen is that amount of radiation that would be absorbed in tissue to the extent of 100 ergs

3-54 EXPOSURE STANDARDS AND PROTECTION REGULATIONS

per gram (mrhm. is an abbreviation for milliroentgens per hour at 1 meter (39.3 inches)).

(i) Liquid radioactive materials Groups I, II or III must, in addition, be packed in tight glass, earthenware, or other suitable inside containers. The inside containers must be surrounded on all sides and within the shield by an absorbent material sufficient to absorb the entire liquid contents and of such nature that its efficiency will not be impaired by chemical reaction with the contents. If the container is packed in a metal container specification 2R or other container approved by the Bureau of Explosives, the absorbent cushioning is not required.

(j) Radioactive materials Group III, liquid, solid, or gaseous, must be packed in suitable inside containers completely wrapped and/or shielded with such material as will prevent the escape of primary corpuscular radiation to the exterior of the shipping container, and secondary radiation at the surface of the container must not exceed 10 milliroentgens per 24 hours, at any time during transportation.

(k) In determining compliance with requirements of paragraphs (e), (g), (h), and (j) of this section measurements of radiation must be made with a Landsverk-Wollan Electrometer Model L-100 or equally efficient standardized meter.

Acceptable instruments for measuring the gamma radiation from packages include the gamma survey meters listed in the SIC series of the AEC Instruments Catalog. Acceptable instruments for the measurement of fast neutron emission include the fast neutron instruments listed in the SIC and SPC Series of the AEC Instruments Catalog.

73.394. *Radioactive materials labels:*

(a) Each outside container of radioactive material Group I or II, unless exempt by 73.392 of this part, must be labeled with a properly executed label as described in 73.414 (a) of this part.

A properly executed label for Group I or Group II requires the name of the principal radioisotope or radioelement, the activity of the contents (measured in curies or in disintegrations per second), the number of units of radiation from the package, and the name of the shipper. If the nature of the radioactive content cannot be appropriately designated by entering a single radioisotope or radioelement, it may be described as "Chemical NOS" (i.e., chemical, not otherwise specified). See comments on paragraph 73.392 (a) (2) for relations between activity and weight of radium, uranium, and thorium.

The number of radiation units is a measure of the gamma radiation from the package as defined in the Note of 73.414 (a), not of the quantity of radioactive material in the package. For cases in which the quantum energy of the radiation is less than 0.15 Mev, see the discussion following paragraph 73.393 (b) above. The actual number of radiation units should be entered on the label. Use of such statements as "less than 10" is unacceptable since this may lead to unnecessary restrictions if several packages occur in the same car, truck, or terminal. (See 75.655 (j) (3) and 77.841 (d) (2) for regulations limiting transportation by one car or vehicle and storage in one place to 40 units.)

(b) Each outside container of radioactive material Group III must unless exempt by 73.392 of this part be labeled with a properly executed "Blue Label."

73.395. *Cleaning cars and vehicles:*

(a) Any box car or motor vehicle which, after use for the transportation of radioactive materials in carload or truckload lots, is contaminated with such materials to the extent that a survey of the interior surface shows that the beta-gamma radiation is greater than 10 milliroentgens physical equivalent in 24 hours or that the average alpha contamination is greater than 500 disintegrations per minute per 100

square centimeters shall be thoroughly cleaned in such a manner that a resurvey of the interior surface shows the contamination to be below these levels. A certificate to that effect must be furnished to the local agent of the carrier or to the driver of the motor vehicle. Cars and motor vehicles which are used solely for the transportation of radioactive materials are exempt from the provisions of this section.

See also paragraph 74.566 (d).

73.396. *Radioactive materials handling:*

(a) When radioactive materials are loaded into railroad cars or motor vehicles by the shipper, the shipper shall observe all applicable requirements of 75.655 (j) or 77.841 (d), as the case may be.

SUBPART H.—MARKING AND LABELING EXPLOSIVES AND OTHER DANGEROUS ARTICLES

73.401. *Dangerous articles:*

(a) Packages containing flammable liquids, flammable solids, oxidizing materials, corrosive liquids, compressed gases, and poisons, as defined in this part must be marked, unless exempted, with the proper shipping name as shown in the commodity list (see 72.5). . . .

* * * * *

(d) Each package must show the specification marking as required if a specification container is prescribed.

(e) Additional shipping information not inconsistent with Parts 71-78 may be shown on a container if so desired but no such label or marking shall be of a design, or form, or size, as may be confused with the marking required by Parts 71-78.

73.402. *Labeling dangerous articles:*

(a) Each package containing any dangerous article as defined in Parts 71-78 must be conspicuously labeled by the shipper as follows, except as otherwise provided:

(1) "Red label" as described in 73.405 on containers of flammable liquids, except when exempted from the regulations by 73.118. If flammable liquid is also a class A poison or a radioactive material poison D, the "poison gas" label or "radioactive materials" label must also be applied to the package.

(2) "Yellow label" as described in 73.406 on containers of flammable solids and oxidizing materials, except when exempted from the regulations by 73.153 and 73.183. If flammable solid or oxidizing material is also a class A poison or radioactive material, the "poison gas" label or "radioactive materials" label must also be applied to the package.

(3) "White label" as described in 73.407 (a) (1), (2) and (3) on containers of acids, alkaline caustic liquids or corrosive liquids, except when exempted from regulations by 73.244. If the acid, alkaline caustic liquid or corrosive liquid is also a class A poison or radioactive material, the "poison gas" label or "radioactive materials" label must also be applied to the package.

(4) "Red label" as described in 73.408 (a) (1) on containers of flammable compressed gases, except when exempted from the regulations by 73.302. If the flammable compressed gas is also a class A poison or radioactive material, the "poison gas" label or "radioactive materials" label must also be applied to the package.

(5) "Green label" as described in 73.408 (a) (2) on containers of nonflammable compressed gases, except when exempted from the regulations by 73.302. If the nonflammable compressed gas is also a class A poison or radioactive material, the

3-56 EXPOSURE STANDARDS AND PROTECTION REGULATIONS

"poison gas" label or "radioactive materials" label must also be applied to the package.

(6) "Poison gas" label as described in 73.409 (a) (1) on containers of class A poisons.

(7) "Poison" label as described in 73.409 (a) (2) on containers of class B poison liquids or solids, except when exempted from the regulations by 73.345 and 73.364. If the class B poison liquid or solid is also radioactive material, the "radioactive materials" label must also be applied to the package.

(8) "Radioactive Materials" label as described in 73.414 (a) on containers of Radioactive Materials Groups I and II except when exempted by 73.392.

(9) "Radioactive Materials" label as described in 73.414 (b) on containers of Group III, except when exempted by 73.392.

* * * * *

(12) "Empty label" as described in 73.413 of this part must be applied to containers which have been emptied and on which the old label has not been removed, obliterated, or destroyed. It must be so placed on the container as to completely cover the old label.

* * * * *

73.403. *Labels for mixed packing:*

(a) Use red label only when red and other labels are prescribed, except when poison gas label or radioactive materials label are prescribed—then both the red label, the poison gas label, or red label and radioactive materials label must be used.

(b) Use white acid (alkaline caustic liquid or corrosive liquid) label only when white acid (alkaline caustic liquid or corrosive liquid) and yellow or poison labels are prescribed or poison labels (class B) are prescribed, except when poison gas label or radioactive materials label are prescribed, then both the white acid label and the poison gas label or white acid and radioactive materials label must be used.

(c) Use yellow label only when yellow and poison labels are prescribed except when poison gas label or radioactive materials label are prescribed then both the yellow label and the poison gas label or the yellow label and the radioactive materials label must be used.

73.404. *Labels:*

(a) Shippers must furnish and attach the labels prescribed for their packages. Labels should be applied to that part of the package bearing consignee's name and address.

(b) Labels must not be applied to packages containing articles which are not subject to Parts 71-78 or are exempted therefrom.

(c) Shippers must not use labels which by their size, shape, and color, may readily be confused with the standard caution labels prescribed in this part.

(d) Labels must conform to standards as to size, printing, and color, and samples will be furnished, on request, by the Bureau of Explosives.

(e) A combination diamond-shaped label-tag of proper size and color, bearing on one side the shipping information and on the reverse side the wording prescribed in this part, will be permitted.

(f) As certification of compliance with regulations is also required by other Government agencies, and to avoid multiplicity of certifications, there may be added to the certificate on labels "and the Commandant of the Coast Guard," or "and the Civil Air Regulations," or "and the Post Office Department," as is necessary.

(g) The carrier's name and stationery form number, or the shipper's name and address, may be printed on the labels, in type not larger than 10 point, if placed within the black-line border and in the upper or lower corner of the diamond.

73.414. *Radioactive materials labels:*

(a) Labels for radioactive materials Group I and Group II must be of diamond shape, white in color, and with each side 4 inches long. Printing must be in red letters inside of a red-line border measuring $3\frac{1}{2}$ inches on each side, as shown in this section. See Fig. 3-1.



Fig. 3-1. Label for Group I and Group II radioactive materials. Red letters on white background.

NOTE.—This label must be duly executed by the shipper and the number of radiation units must be shown. For purposes of these regulations 1 unit equals 1 milliroentgen per hour at 1 meter for hard gamma radiation or the amount of radiation which has the same effect on film as 1 mrhm of hard gamma rays of radium filtered by $\frac{1}{2}$ inch of lead.

(b) Labels for radioactive materials (class D poisons Group III) must be of diamond shape, white in color, and with each side 4 inches long. Printing must be in blue letters inside of a blue-line border measuring $3\frac{1}{2}$ inches on each side, and as shown in this section. See Fig. 3-2.



Fig. 3-2. Label for Group III radioactive material. Blue letters on white background.

73.427. Shipping order and bill of lading description:

(a) The shipper when offering for transportation by carriers by rail freight, rail express, highway, or water any class A, class B, or class C explosive, flammable liquid, flammable solid, oxidizing material, corrosive liquid, compressed gas, or poison, as defined by this part, must describe such article in the shipping order, bill of lading or other shipping paper by the shipping name used in 72.5 (see commodity list) and may add a further description not inconsistent therewith. Abbreviations must not be used. . . .

The commodity list given in section 72.5 indicates that all radioactive materials should be entered on shipping papers as "radioactive materials." The group classification, as defined by paragraph 73.391 (a) should be added to this description.

73.428. Label or placard notation:

(a) The shipping order, bill of lading or other shipping paper must also show thereon in connection with the entry of the article as prescribed in 73.427 of this part, the color or kind of label applied, and for cars containing such articles loaded by the shipper, requiring placards the kind of placard applied to the car.

Labels applied to packages of radioactive material should be identified by color; i.e., as red or blue.

73.430. *Certificate:*

(a) The shipper offering for transportation by carriers by rail freight, highway, water, or air, any class A or class B explosive and blasting caps or electric blasting caps in any quantity, and any flammable liquid, flammable solid, oxidizing material, corrosive liquid, compressed gas, or poison, requiring labels, or carloads requiring placards, as prescribed by Parts 71-78, must show on the shipping order, bill of lading, or other shipping paper, in the lower left-hand corner, the following certificate over the written or stamped facsimile signature of the shipper or his duly authorized agent:

This is to certify that the above-named articles are properly described, and are packed and marked and are in proper condition for transportation according to the regulations prescribed by the Interstate Commerce Commission.

(b) For the relief of shippers from multiplicity of certifications required for packages which may move by various means of transportation, shipments may be certified for rail, motor vehicle, water, or air transportation by adding to the certificate required on the shipping document "and the Commandant of the Coast Guard," or "and the Civil Air Regulations," as the case may be.

PART 74.—REGULATIONS APPLYING PARTICULARLY TO CARRIERS BY RAIL FREIGHT

SUBPART A.—LOADING, UNLOADING, PLACARDING AND HANDLING CARS; LOADING PACKAGES INTO CARS

73.532. *Loading other dangerous articles into cars:*

(a) Shipments must be properly loaded in closed cars, except as otherwise provided in Parts 71-78 and cars placarded as prescribed, when accepted by carriers.

* * * * *

(j) Radioactive ores, residues, and similar material: Shipment of radioactive ores, residues, or similar material as provided in 73.392, must be so loaded as to avoid spillage and scattering of loose material.

(1) The amount of radioactive material loaded in a car must be limited as provided in 73.392.

(2) No person shall remain in a car containing radioactive material unnecessarily and the shipper must furnish the carrier with such information and equipment as is necessary for the protection of the carrier's employees.

(3) Any loose radioactive material must be removed from the car and placed in a closed container in a segregated location and held for instructions from the shipper or the Bureau of Explosives.

* * * * *

SUBPART C.—PLACARDS ON CARS

74.541. *"Dangerous" placards; "Dangerous—Radioactive Material" placards:*

* * * * *

(b) "Dangerous—Radioactive Material" placards, as prescribed in 74.553 must be applied to cars containing shipments of class D poisons as provided in 73.391 and 73.392.

3-60 EXPOSURE STANDARDS AND PROTECTION REGULATIONS

74.553. *Dangerous—Class D Poison placard:*

(a) The "Dangerous—Radioactive Material" placard for class D poisons, must be of diamond shape measuring $10\frac{3}{4}$ inches on each side, and must bear the wording in red letters as shown in Fig. 3-3.

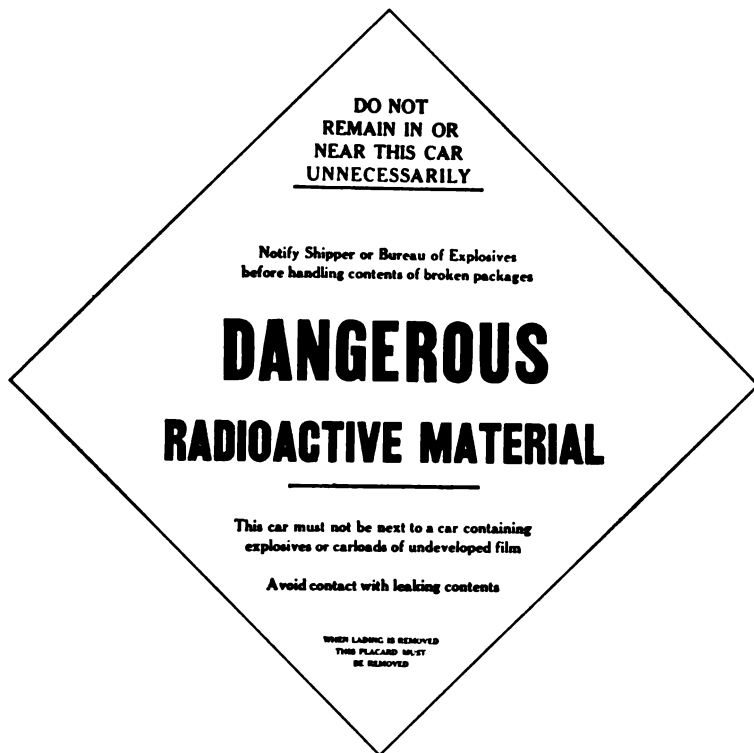


FIG. 3-3. Placard for radioactive material. Red letters on white background.

74.554. *Unauthorized use of placards:*

(a) Placards prescribed by this part must not be applied to cars containing articles not subject to Parts 71-78 or specifically exempted therefrom.

SUBPART D.—UNLOADING FROM CARS

74.566. *Cleaning cars:*

* * * * *

(d) Any box car which, after use for the transportation of radioactive materials in carload lots, is contaminated with such materials to the extent that a survey of the interior surface shows that the beta-gamma radiation is greater than 10 milliroentgens physical equivalent in 24 hours or that the average alpha contamination is greater than 500 disintegrations per minute per 100 square centimeters shall be thoroughly cleaned in such a manner that a resurvey of the interior surface shows the contamination to be below these levels. A certificate to that effect must be furnished to the local agent of the carrier. Cars which are used solely for the transportation of radioactive materials are exempt from the provisions of this section.

74.597. *Leaking packages of acid or poisons:*

* * * * *

(e) *Radioactive materials.—Poison Class D.*—In event of breakage of container, wreck, fire, or unusual delay involving cars placarded “Dangerous—Radioactive Material” as prescribed in 74.541 (b) of this part, the car and any loose radioactive material must be isolated as far as possible from danger of human contact and no persons must be allowed to remain close to the car or contents needlessly until qualified persons are available to supervise handling. The shipper and the Bureau of Explosives should be notified immediately.

(1) Cars, buildings, area, or equipment in which class D poisons have been spilled must not be again placed in service or occupied until decontaminated by qualified persons.

PART 76.—REGULATIONS APPLYING TO RAIL CARRIERS IN BAGGAGE SERVICE

76.702. *Dangerous articles:*

(a) No dangerous article described by Parts 71-78 shall be accepted for transportation or transported in rail baggage service except as provided for in 76.703 of this part. . . .

The list of acceptable articles given in 76.703 does not include radioactive materials.

PART 77.—REGULATIONS APPLYING TO SHIPMENTS MADE BY WAY OF COMMON, CONTRACT, OR PRIVATE CARRIERS BY PUBLIC HIGHWAY

SUBPART A.—GENERAL INFORMATION AND REGULATIONS

77.802. *Application of regulations in Parts 71-78:*

(a) Parts 71-78 apply to all common and contract carriers by motor vehicle transporting explosives and/or other dangerous articles as defined by Interstate Commerce Commission “Regulations for Transportation of Explosives and Other Dangerous Articles by Land and Water in Rail Freight, Express and Baggage Services and by Motor Vehicle (Highway) and Water.” When shipments are accepted by motor vehicle for further transportation by rail express . . . , rail freight or by water on board vessel, they must in addition to Parts 71-78, comply with the applicable regulations for the service by which they are to be transported.

* * * * *

77.815. *Labels:*

(a) Labels prescribed by the Commission’s regulations, Part 73, must have been applied to shipments, unless exempt from Parts 71-78, and in addition the shipper must have certified to compliance with the regulations by writing, stamping, or printing his name underneath the certificate printed thereon or on the shipping papers.

* * * * *

77.823. *Marking on motor vehicles and trailers other than tank motor vehicles:*

(a) Except as otherwise provided in Part 73, every motor vehicle transporting explosives or other dangerous articles must be marked or placarded as follows:

* * * * *

(3) Every motor vehicle transporting any quantity of radioactive material, class D poison, requiring red radioactive materials label must be marked or placarded

3-62 EXPOSURE STANDARDS AND PROTECTION REGULATIONS

"DANGEROUS—RADIOACTIVE MATERIALS" on each side and rear with a placard or lettering in letters not less than 3 inches high on a contrasting background.

(4) Every motor vehicle transporting 2,500 pounds gross weight or more of explosives class B, flammable liquids, flammable solids, oxidizing materials, acids or other corrosive liquids, compressed gases (see Note 1), class B poisons, class C poisons, and class D poisons not requiring red radioactive materials label must be marked or placarded "DANGEROUS" on each side and rear with a placard or lettering in letters not less than 3 inches high on a contrasting background: Provided, however, that if such articles are, because of size and kind of containers, exempted from the packaging, marking, and labeling requirements of Part 73, and provided such exempted commodities do not have a gross weight (contents and containers) exceeding 5,000 pounds, the provisions of this subparagraph shall not be applicable. . . .

SUBPART B.—LOADING AND UNLOADING

77.841. *Poisons:*

* * * * *

(d) *Radioactive material.*—A container of radioactive material bearing radioactive material, red label, must not be placed in vehicles, terminals, or other places closer than 3 feet to an area which may be continuously occupied by passengers, employees, or shipments of animals. When more than one such container is present, the distance from occupied areas must be computed from the table in subparagraph (1) of this paragraph by adding the number of units shown on labels on the containers.

* * * * *

(2) Not more than 40 units of radioactive material, red label, shall be transported in any vehicle or stored in any location at one time. Packages must be so blocked or braced in vehicles as to prevent any shift of lading under conditions normally incident to transportation.

* * * * *

SUBPART D.—VEHICLES AND SHIPMENT IN TRANSIT: ACCIDENTS

77.860. *Accidents; poisons:*

(d) *Cleaning vehicles.*—Any motor vehicle which, after use for the transportation of radioactive materials in truckload lots, is contaminated with such materials to the extent that a survey of the interior surface shows that the beta-gamma radiation is greater than 10 milliroentgens physical equivalent in 24 hours or that the average alpha contamination is greater than 500 disintegrations per minute per 100 square centimeters shall be thoroughly cleaned in such a manner that a resurvey of the interior surface shows the contamination to be below these levels. A certificate to that effect must be furnished the carrier or to the driver of the motor vehicle. Motor vehicles which are used solely for the transportation of radioactive materials are exempt from the provisions of this section.*

* Revised May 3, 1955.

SUBPART E.—REGULATIONS APPLYING TO EXPLOSIVES OR OTHER DANGEROUS ARTICLES
ON MOTOR VEHICLES CARRYING PASSENGERS FOR HIRE77.870. *Regulations for passenger-carrying vehicles:*

* * * * *

(g) *Radioactive materials on passenger-carrying vehicles.*—No motor carrier may transport any radioactive material, poison, class D, requiring red or blue radioactive material label under these regulations in or on any bus while engaged in the transportation of passengers except where no other practicable means of transportation is available. Packages of radioactive materials must be handled and placed in the vehicle in accordance with the requirements of 77.841 (d) of this part.

CIVIL AIR REGULATIONS GOVERNING THE TRANSPORTATION OF
RADIOACTIVE MATERIALS

The transportation of radioactive materials by air is regulated by the Civil Aeronautics Board. Its **Civil Air Regulations** covering the Transportation of Explosives and Other Dangerous Articles constitutes Part 49 of Title 14 of the Code of Federal Regulations. Those regulations of particular interest to shippers of radioactive materials are excerpted below.

Many of the Civil Air Regulations applying to the shipment of radioactive materials—especially those applying to packaging, marking, or labeling of such materials—are based on ICC regulations. Comments made in Chapter II on ICC regulations are not repeated here. The few comments made in this section are distinguished from the text of the excerpts by the comments not being indented. Limitations on the nature of these comments are given at the beginning of the ICC comments.

PART 49.—TRANSPORTATION OF EXPLOSIVES AND OTHER DANGEROUS ARTICLES

49.0. *Applicability of part:*

Explosives or other dangerous articles, including flammable liquids, flammable solids, oxidizing materials, corrosive liquids, compressed gases, and poisonous substances, shall not be loaded in or transported by civil aircraft in the United States, or transported anywhere in air commerce in civil aircraft of United States registry except as hereinafter provided.

49.1. *Definitions:*

(a) As used in this part the words listed below shall be defined as follows:

7 (iv) *Radioactive materials—Class D.*—A radioactive material is any material or group of materials which spontaneously emits ionizing radiation. For the purpose of these rules, radioactive materials are divided into three groups according to the type of radiation emitted at any time during transportation as follows:

(I) *Group I radioactive materials.*—Group I radioactive materials are those materials which emit any gamma radiation, either alone or with electrically charged particles or corpuscles.

(II) *Group II radioactive materials.*—Group II radioactive materials are those materials which emit neutrons and either or both of the types of radiation characteristic of Group I radioactive materials.

3-64 EXPOSURE STANDARDS AND PROTECTION REGULATIONS

(III) *Group III radioactive materials.*—Group III radioactive materials are those materials which emit only electrically charged particles or corpuscles (i.e., alpha and/or beta radiation).

The above definitions of Group I and Group III materials differ slightly from the ICC definitions. See 73.391 (a).

(8) "Unit" of gamma radiation. "Unit" of gamma radiation is one milliroentgen per hour at a meter for "hard gamma" radiation, i.e., that amount of gamma radiation which will have the same effect on sensitive photographic film as one milliroentgen per hour at a meter of "hard gamma" radiation of radium filtered through $\frac{1}{2}$ inch of lead.

(10) Cargo aircraft. A cargo aircraft is an aircraft other than a passenger-carrying aircraft which is carrying goods or property.

(11) Marking. Marking is the display on the container of the name of the articles inside as listed in the commodity list of the ICC Regulations.

(12) Labeling. Labeling is the display on the container of an appropriate label as specified for a particular class of articles by the ICC Regulations.

(13) ICC Regulations. ICC Regulations shall mean the "Interstate Commerce Commission's Regulations for Transportation of Explosives and Other Dangerous Articles," effective January 7, 1941, as amended or revised from time to time.

(14) Aircraft operator. An operator of aircraft shall include the owner, lessee, or any other person who causes or authorizes the operation of the aircraft.

49.3. *Packing, marking, and labeling requirements:*

(a) Unless otherwise specifically provided in this part, explosives or other dangerous articles shipped by air shall be packed, marked, and labeled in accordance with the specifications established in Part 72 * of the ICC Regulations for transportation by rail express: *Provided*, That liquids shall be packed only in containers which are securely closed, sufficient strength to prevent any leakage or distortion of the containers caused by change in temperature or altitude during transit, and so filled as to provide adequate outage. All explosives or other dangerous articles shipped by air shall show the proper shipping name as shown in the commodity list of Part 72 of the ICC Regulations and any instructions that are necessary for safe handling.

(b) No shipper shall offer and no air carrier or other operator of aircraft shall knowingly accept explosives or dangerous articles for carriage

by air unless the shipper or his authorized agent has certified that the shipment complies with the requirements of this part. No shipment shall be accepted for transportation by passenger-carrying aircraft unless the package shows a clear and plainly visible statement that it is within the limitations prescribed for passenger operations. Any operator of aircraft may rely on such a certificate as prima facie evidence that the shipment so certified complies with the requirements of this part.†

* Part 72 of the ICC Regulations incorporates the packaging specifications of Part 73 thereof. It will be noted that items exempted from the packaging, labeling, or marking provisions of Part 73 of the ICC Regulations are *not* exempted from such requirements for shipment by air unless it is expressly so provided in this part.

† The following statement on a shipping label signed by a responsible agent of the shipper will be accepted as meeting this requirement: This is to certify that the contents of this package are properly described by name and are packed and marked and are in proper condition for transportation according to the regulations prescribed by the Interstate Commerce Commission and the Civil Aeronautics Board.

For shipment on passenger-carrying aircraft add the following: This shipment is within the limitations prescribed for passenger-carrying aircraft.

The first of the statements given in the footnote below appears on the radioactive material labels prescribed by the Interstate Commerce Commission, paragraphs 73.414 (a) and (b).

PASSENGER-CARRYING AIRCRAFT

49.10. *Acceptable explosives and other dangerous articles on aircraft carrying passengers:* No article listed in Appendices A or B of this part shall be carried on passenger-carrying aircraft, and no other explosives or dangerous article shall be carried in passenger-carrying aircraft except as provided in paragraphs 49.11 through 49.18.

49.18. *Radioactive materials:* Radioactive materials—Class D, Groups I, II, and III (liquid, solid, or gaseous) may be carried when packed, marked, and labeled in accordance with the provisions of paragraphs 73.368 through 73.369 of the ICC Regulations. (See paragraph 49.55 for handling of radioactive materials in aircraft. See also paragraph 49.62 where certain other types of radioactive materials are exempted from certain of the requirements of this part.)

As the ICC Regulations are now numbered, the paragraphs containing provisions for packing, marking, and labeling of radioactive materials are 73.391 through 73.394.

CARGO AIRCRAFT

49.41. *Articles which may be carried in cargo aircraft:* In addition to the articles acceptable for transportation on aircraft carrying passengers, any article acceptable for, and packed, marked, and labeled in accordance with the ICC Regulations for transportation by rail express may be carried in cargo aircraft: Provided, That no article listed in Appendix A of this part shall be carried except under the provisions of 49.71. The maximum quantity in any one outside package or container shall not exceed that prescribed in the commodity list of Part 72 of the ICC Regulations.

The commodity list of Part 72 of the ICC Regulations lists the "Maximum quantity in 1 outside container by rail express" for radioactive materials as follows: "2000 milluries. See 73.391 (c)."

EXEMPTED ARTICLES

49.62. *Radioactive materials:*

(a) Radioactive materials which meet all of the following conditions are exempt from packing, marking, and labeling requirements required by this part:

(1) The package shall be such that there can be no leakage of radioactive material under conditions normally incident to transportation.

(2) The package shall contain not more than 0.1 millicurie of radium, or polonium, or that amount of strontium 89, strontium 90, or barium 140 which disintegrates at a rate of more than 5 million atoms per second; or not more than that amount of any other radioactive substance which disintegrates at a rate of more than 50 million atoms per second.

(3) The package shall be such that no significant alpha, beta, or neutron radiation is emitted from the exterior of the package, and the gamma radiation at any surface of the package shall be less than 10 milliroentgens in 24 hours.

(b) Manufactured articles other than liquids, such as instrument or clock dials of which radioactive materials are a component part, and luminous compounds when securely packed in strong outside containers are exempt from packing, marking, and labeling requirements, provided the gamma radiation at any surface of the package is less than 10 milliroentgens in 24 hours.

(c) (1) Radioactive materials, such as ores, residues, etc., packed in strong, tight containers are exempt from packing and labeling requirements for shipment in plane-load lots, provided the per-planeload (as loaded in place in the airplane) does not exceed 10 milliroentgens per hour of gamma radiation or equivalent. There shall be no loose radioactive material in the airplane, and the shipment must be braced and lashed so as to prevent leakage or shift of lading under normal conditions of flight.

(2) It is the responsibility of the consignor and/or consignee to supervise, respectively, all loading and unloading operations and to monitor all personnel involved so that the accepted limits of personnel radiation exposure are not exceeded.

(d) Shipments of radioactive materials made by the Atomic Energy Commission or under its direction or supervision, which are escorted by personnel who are specially designated by the Atomic Energy Commission, are exempted from the provisions of these regulations where special arrangements are made with and approved by the Administrator.

See elsewhere for discussion of a similar provision in the ICC Regulations, 73.7 (b). Note also that although ICC Regulations do not require special arrangement for such exemption, Civil Air Regulations do require such arrangements.

49.71. *Special Authority:*

In emergency situations or where other forms of transportation are impracticable:

(a) Deviations * from any of the provisions of this part for a particular flight may be authorized by the Administrator where he finds that the conditions under which the articles are to be carried are such as to permit the safe carriage of persons and cargo.

(b) Deviations from the 2,000 millicurie quantity limitation prescribed for radioactive materials by paragraph 49.18 may be taken by the Atomic Energy Commission for Atomic Energy Commission shipments, provided that such shipments are made in accordance with the requirements approved by the Interstate Commerce Commission for shipment by rail express and prior notification of each shipment is given by the Atomic Energy Commission in the form and manner prescribed by the Administrator.

U.S. COAST GUARD REGULATIONS GOVERNING THE TRANSPORTATION OF RADIOACTIVE MATERIALS

The following regulations of the **U.S. Coast Guard** governing the packaging, marking, and labeling of radioactive materials transported by water are excerpted from Title 46, Part 146, of the Code of Federal Regulations. It will be observed that they are in very close agreement with the corresponding ICC Regulations.

PART 146.—REGULATIONS OF THE U.S. COAST GUARD COVERING THE TRANSPORTATION OR STORAGE OF EXPLOSIVES OR OTHER DANGEROUS ARTICLES OR SUBSTANCES AND COMBUSTIBLE LIQUIDS ON BOARD VESSELS

146.02-8. *U.S. Government shipments:*

(b) Shipments of radioactive materials, made by the Atomic Energy Commission, or under its direction or supervision, which are escorted by personnel specially designated by the Atomic Energy Commission, are exempt from the regulations in this part.

* The CAB has expressed the opinion to the AEC (informal telephone opinion May 29, 1957) that a shipment under a Bureau of Explosives Permit is not considered a deviation from the provisions of this part.

146.05-15. *Marking and labeling applying to domestic shipments only:*

(g) Each package containing "Any Other Dangerous Article" as defined by the regulations in this part shall be conspicuously labeled by the shipper as follows, except as otherwise provided:

(11) "Radioactive materials label" as described and illustrated in paragraph 146.05-17 (g) on containers of Group I and Group II radioactive materials.

(12) "Radioactive materials label" as described and illustrated in paragraph 146.05-17 (r) on containers of Group III radioactive materials.

146.05-17¹. *Labels:*

(a) Shippers shall furnish and attach the labels prescribed for their packages.

(d) Labels shall conform to standard as required by Interstate Commerce Commission regulations.

(e) A combination diamond-shaped label-tag of proper size and color, bearing on one side the shipping information and on the reverse side the wording prescribed in this section, will be permitted.

SUBPART E.—DETAILED REGULATIONS GOVERNING POISONOUS ARTICLES

146.25-1. *Definition of poisonous articles:*

(a) Poisonous articles are divided by the Interstate Commerce Commission regulations into four classes according to degree of hazard in transportation. These are:

Extremely dangerous poisons—Class A.

Less dangerous poisons—Class B.

Tear gases or irritating substances—Class C.

Radioactive materials—Class D.

(b) These poisonous articles are defined by the Interstate Commerce Commission regulations as set forth in paragraphs 146.25-5, 146.25-10, 146.25-15, and 146.25-20, and such definitions are binding upon all shippers making shipments of poisonous articles by common carrier vessels engaged in interstate or foreign commerce by water. These definitions are accepted and adopted and form part of the regulations in this subchapter and apply to all shippers making shipments of poisonous articles by any vessel and shall apply to owners, charterers, agents, master or other person in charge of a vessel, and to other persons, transporting, carrying, conveying, storing, stowing, or using poisonous articles on board vessels subject to R.S. 4472, as amended (46 U.S.C. 170), and the regulations in this subchapter.

146.25-20. *Radioactive materials, Class D, radioactive materials label:*

(a) Radioactive material is any material or combination of materials that spontaneously emits ionizing radiation. For the purpose of the regulations in this part radioactive materials are divided into 3 groups according to the type of rays emitted at any time during transportation, as follows:

(1) *Group I.*—Radioactive materials that emit gamma rays only or both gamma and electrically charged corpuscular rays.

(2) *Group II.*—Radioactive materials that emit neutrons and either or both types of radiation characteristic of Group I materials.

(3) *Group III.*—Radioactive materials that emit electrically charged corpuscular rays only, i.e., alpha or beta, etc., or any other radioactive material that

3-68 EXPOSURE STANDARDS AND PROTECTION REGULATIONS

is so shielded that the gamma radiation at the surface of the package does not exceed 10 milliroentgens * per 24 hours at any time during transportation.

146.25-25. *Exemptions for radioactive materials:*

(a) Radioactive materials are exempt from prescribed packaging, marking other than the name of the contents, and labeling requirements, provided they fulfill all of the following conditions:

(1) The package must be such that there can be no leakage of radioactive materials under conditions normally incident to transportation.

(2) The package must contain not more than 0.1 millicurie † of radium, or polonium or that amount of strontium 89, strontium 90, or barium 140 which disintegrates at a rate of more than 5 million atoms per second; or that amount of any other radioactive substance which disintegrates at a rate of more than 50 million atoms per second.

(3) The package must be such that no significant alpha, beta, or neutron radiation is emitted from the exterior of the package, and the gamma radiation at any surface of the package must be less than 10 milliroentgens per 24 hours.

(b) Manufactured articles other than liquids, such as instrument or clock dials of which radioactive materials are a component part, and luminous compounds, when securely packed in strong outside containers are exempted from specification packaging, marking other than name of contents, and labeling requirements provided the gamma radiation at any surface of the package is less than 10 milliroentgens per 24 hours.

(c) Radioactive materials, such as ores, residues, etc., of low activity, packed in strong tight containers, are exempt from specification packaging, marking other than name of contents, and labeling requirements for transportation on board vessels only if the gamma radiation or equivalent at any point in any space or area continuously occupied by passengers, crew, or shipments of animals, will not exceed 40 milliroentgens per 24 hours at any time during transportation.

146.25-30. *Packing and shielding of radioactive materials:*

(a) Not more than 2,000 millicuries of radium, polonium, or other members of the radium family of elements, and not more than that amount of any other radioactive substance which disintegrates at a rate of 100,000 million (10^{11}) atoms per second may be packed in one outside container for transportation on board vessels, except by special arrangements and under conditions approved by the Commandant of the Coast Guard.

(b) Radioactive materials that present special hazards due to their tendency to remain fixed in the human body for long periods of time (i.e., radium, plutonium, and radioactive strontium, etc.) must, in addition to the packing prescribed in this subpart, be packed inside metal containers (ICC specification 2R) or other containers approved by the Bureau of Explosives, and authorized by the Commandant of the Coast Guard.

(c) All radioactive materials must be so packed and shielded that the degree of fogging of undeveloped film under conditions normally incident to transportation (24 hours at 15 feet from the package) will not exceed that produced by 11.5 milliroentgens of penetrating gamma rays of radium filtered through $\frac{1}{2}$ inch of lead.

* In determining compliance with requirements of these regulations, all measurements of radiation must be made with a Landsverk-Wollan Electrometer Model L-100, or equally efficient standardized meter.

† For purposes of the regulations in this part 1 millicurie is that amount of any radioactive material which disintegrates at the rate of 37 million atoms per second.

(d) The design and preparation of the package must be such that there will be no significant radioactive surface contamination of any part of the container.

(e) The smallest dimension of any outside shipping container for radioactive materials must not be less than 4 inches.

(f) All outside shipping containers must be of such design that the gamma radiation will not exceed 200 milliroentgens per hour or equivalent at any point of readily accessible surface. Containers must be equipped with handles and protective devices when necessary in order to satisfy this requirement.

(g) The outside shipping container for radioactive material, unless specifically exempt by paragraph 146.25-25 (a), (b), or (c), must be a wooden box (ICC specification 15A or 15B), a fiber drum (ICC specification 21A), or a fiberboard box (ICC specification 12B), except that equally efficient containers may be used when approved by the Bureau of Explosives and authorized by the Commandant of the Coast Guard.

(h) Radioactive materials, Group I, liquid, solid, or gaseous, must be packed in suitable inside containers completely surrounded by a shield of lead or other suitable material of such thickness that at any time during transportation the gamma radiation will not exceed 10 milliroentgens per hour at a distance of one meter (39.3 inches). The shield must be so designed that it will not break or open under conditions incident to transportation. The minimum shielding must be sufficient to prevent the escape of any primary corpuscular radiation to the exterior of the outside shipping container.

(j) (1) Radioactive materials, Group II, liquid, solid, or gaseous, must be packed in suitable inside containers completely shielded so that at any time during transportation the radiation measured at right angles to any point on the long axis of the shipping container will not exceed the following specified limits:

(i) Gamma radiation of 10 mrhm.

(ii) Electrically charged corpuscular radiation which is the physical equivalent * of 10 mrhm. of gamma radiation.

(iii) Neutron radiation which is the physical equivalent of 2 mrhm. of gamma radiation.

(iv) If more than one type of radiation named in subdivisions (i), (ii), and/or (iii) of this subparagraph is present, the radiation of each type must be reduced by shielding so that the total does not exceed the equivalent of subdivisions (i), (ii), or (iii) of this subparagraph.

(2) The shielding must be designed so as to maintain its efficiency under conditions normally incident to transportation and must provide personnel protection against fast or slow neutrons and all other ionizing radiation originating in the radioactive materials or any part of the aggregate constituting the complete package.

* For the purposes of the regulations in this part the "Physical Equivalent" of a roentgen is that dose of any ionizing radiation which results in the absorption in tissue of ionizing energy equivalent to 93 ergs per gram of tissue. This is approximately the dose which is imparted to soft tissue by 1 roentgen of gamma or X-rays but it may be imparted by corpuscular radiation which is not measured in terms of roentgens.

U.S. POSTAL REGULATIONS COVERING THE TRANSPORTATION OF
RADIOACTIVE MATERIALS

The following **U.S. Postal Regulations** covering the transportation of radioactive materials by mail are excerpted from Part I of the U.S. Postal Guide, Article 37, Chapter IV, 1951.

37. *Radioactive materials:*

(a) Radioactive materials requiring a caution label under Interstate Commerce Commission regulations are prohibited. But certain of these materials which are exempt from specification packing, marking, and labeling requirements under Interstate Commerce Commission regulation 73.392 Explosives and Other Dangerous Articles, may be admitted to the mails if packed and labeled in accordance with paragraphs (b) to (f).

All prospective mailers of these materials are cautioned that mailings must be within the amount specified and postmasters shall assure themselves of the fact that the mailer is in a position to determine definitely the radiation of any proposed mailing before it is accepted.

(b) Radioactive materials (liquid, solid, or gaseous; manufactured articles such as instrument or clock dials of which radioactive materials are a component part; luminous compounds; ores and residues) which fulfill all the following conditions shall be accepted for mailing provided they are packed in a strong, tight outside container and marked "Radioactive Material—Gamma Radiation at Surface of Parcel Less Than 10 Milliroentgens for 24 Hours—No Significant Alpha, Beta, or Neutron Radiation."

(c) The package must be such that there can be no leakage of radioactive material under conditions normally incident to transportation in the mails in sacks.

(d) The package must contain not more than 0.1 millicurie of radium, or polonium, or that amount of strontium 89, strontium 90, or barium 140 which disintegrates at a rate of more than 5 million atoms per second; or that amount of any other radioactive substance which disintegrates at a rate of more than 50 million atoms per second.

(e) The package must be such that no significant alpha, beta, or neutron radiation is emitted from the exterior of the package and the gamma radiation at any surface of the package must be less than 10 milliroentgens for 24 hours.

(f) The design and preparation of the package of radioactive material must be such that there will be no significant radioactive surface contamination of any part of the container. Liquids must be packed in tight glass, earthenware, or other suitable inside containers surrounded by an absorbent material sufficient to absorb the entire liquid contents and of such nature that its efficiency will not be impaired by chemical reaction with the contents.

NOTE.—The amounts of radioactive materials shown are based on exemptions to Interstate Commerce Commission regulation 73.392, while packaging requirements are based on Interstate Commerce Commission regulation 73.393.

AUXILIARY INFORMATION FOR SHIPPERS

Regulations Pertaining Particularly to Carriers. There are certain sections of the various Regulations * not of direct concern to shippers or consignees. They are of interest,

* All references in this subsection to Regulations refer to the appropriate Interstate Commerce Commission, Civil Aeronautics Board and U.S. Coast Guard Regulations.

however, since they may influence the manner of shipment or involve the shipper or consignee in responsibility, for example:

A. ICC Regulations:

Part 74.—Regulations Applying to Carriers by Rail Freight.

Part 75.—Regulations Applying to Carriers by Rail Express.

Part 76.—Regulations Applying to Rail Carrier in Baggage Service (LCL Movements).

Part 77.—Regulations Applying to Shipments Made by Way of Common, Contract, or Private Carriers by Public Highway (Motor Carriers).

B. Civil Aeronautics Board Regulations:

49.51 through 49.55. Loading and Handling Requirements (Commercial and Contract Air Lines regulated by the C.A.B.).

C. Coast Guard Regulations:

146.02-13 Report Fires.

146.02-14 Damaged Containers.

146.02-15 Emergency Shipments.

146.02-16 Shipments in Violation.

146.20-90 Stowage and Storage Chart of Dangerous Articles.

146.25-35 Stowage and Handling Aboard Vessels.

146.25-45 Limitation on All Stowage.

146.25-50 Care Following Leakage or Sifting of Poisonous Articles.

Limitation on Number of "Units."—Since no carrier by rail express, motor truck, or air is permitted to carry more than 40 units and since surface carriers are prohibited from storing or stowing more than 40 units in any location, consignments should not be offered for shipment that cannot be accepted by a carrier because of such quantity limitations. (See ICC 75.655 (j) (3); 77.841 (d) (2); CAB 49.54 (c); USCG 146.25-45 (g).)

Accident, Leakage, Spillage, or Sifting.—In cases of accident, leakage, spillage, or sifting, the carrier is usually required to notify the shipper and/or the proper authorities. The shipper may be required to render assistance in removing or disposing of the material and in decontaminating equipment or property. (See, for example, ICC 74.532 (j) (3); 75.655 (j) (7); 77.860 (c); CAB 49.55 (a) and (b); USCG 146.25-50 (b).)

Aircraft Engaged Principally in Transportation of Radioactive Materials.—If any aircraft is engaged principally or entirely in the transportation of radioactive materials, it is the responsibility jointly of the shipper and the carrier to monitor all personnel involved so that the accepted limits of personnel radiation exposure are not exceeded (CAB 49.55 (d).)

Shipments of Radioactive Materials.—The shipper should notify the receiver of any shipment of radioactive materials, giving the following information: (1) date of shipment, (2) expected date of arrival, (3) method of transportation, and (4) description of material, packaging, etc. This information should be sent to reach the consignee before or at the time of arrival of the shipment.

It is suggested that shippers will find it advantageous to keep complete records of all shipments covered by these regulations. Such records should cover the following items: (1) description and quantities of material; (2) radiation measurements and instrumentation used; and (3) labels or placards affixed.

Violations of Regulations.—Shipments received by the consignee which appear to have been made in violation of regulations should be reported by consignee through normal contractual channels. (See ICC 73.11.)

3-72 EXPOSURE STANDARDS AND PROTECTION REGULATIONS

Permits.—The several regulatory agencies can, within their discretion, issue special permits to cover shipments which do not meet the requirements of their regulations. AEC contractors, planning shipments not covered by the regulations or the use of shipping containers not provided for in the regulations, should submit a permit application through regular contractual channels to the Operations Office.

The application should include the following information:

Consignee and destination.

Method of shipment.

Description of material and quantity.

Date.

Gross weight of material.

Type of container (a drawing, sample or model should be submitted in the case of a new type of container for which a permit is being requested).

Radiation—Group classification (I, II, or III).

A. At surface.

B. Radiation units.

Labels used.

Section 4

NATURAL RADIOACTIVE BACKGROUND

By

FREDERICK P. COWAN, Ph.D., *Head of Health Physics Division, Brookhaven National Laboratory.*

General Nature of Background.....	4-2
Cosmic Radiation.....	4-3
Radioactivity of the Earth's Crust.....	4-7
Radioactivity of Water.....	4-11
Radioactivity of Air.....	4-12
Radioactivity of the Human Body.....	4-15

NATURAL RADIOACTIVE BACKGROUND

Frederick P. Cowan

GENERAL NATURE OF BACKGROUND

The term **natural radiation background** is used to designate naturally occurring radioactive materials and high-energy radiations. The naturally occurring radioactive materials are described in detail in Sec. 6. The various members of the uranium and thorium families and a radioactive isotope of potassium are the most important. Naturally occurring radiations are due partly to these natural radioactive materials and partly to the cosmic radiation which is described in detail below.

The **total background radiation** levels to which people may be exposed are of considerable interest. Measurements are usually made with ionization chambers and the results expressed in milliroentgens per year, i.e., in terms of the rate of energy absorption. The sea-level value is about 0.01 mr/hr (88 mr/year) in regions of low background but will be considerably higher in many places. At higher elevations the total background increases sharply because of the increased contribution from cosmic rays and the presence of local mineral deposits. Values inside buildings are often higher than outdoors because of the effect of the building materials. Some representative values of total background are given in Tables 4-1 and 4-2. Extensive measurements of various aspects of natural background radiation have been made at a number of laboratories. (Sedlet, J., Environmental Radioactivity at

Table 4-1. Total Radiation Dosages from Normal Background Radiation, mr/year *

Altitude of ground surface, ft	Ordinary granite		Typical sedimentary rock		Open ocean	
	Equator	55°N	Equator	55°N	Equator	55°N
Sea level	143	147	76	80	53	57
5,000	150	170	83	103		
10,000	190	230	123	163		
15,000	270	350	203	283		
20,000	414	560	347	493		

* From Libby, W. F., Dosages from Natural Radioactivity and Cosmic Rays, *Science*, vol. 122, no. 3158, pp. 57-58, July 8, 1955.

Argonne National Laboratory—Report for the Year 1956, *Rept.* ANL-5808, January, 1958. Blifford, I. H., *et al.*, Radioactivity of the Air, *Naval Research Lab. Rept.* NRL-4760, June, 1956.)

Table 4-2. Experimental Data for Hard Background Radiation, mr/year *

Observer	Cosmic rays	Gamma rays		Cosmic and gamma rays (total)	Location
		From air	From ground		
Sievert and Hultquist	44	...	...	121-150	Streets of Stockholm Over igneous rocks, Sweden Clay soil Wood houses (average center of room) Brick and concrete houses (types 1, 2) Brick and concrete houses (type 3)
				104-182	
				94	
				104	
				145	
Cowan.....		...	...	296	Outdoors, Brookhaven, N.Y., measured
				(max 520)	
Hess and Vancour...	34	2	53	90	Outdoors, Fordham University campus, N.Y., 1 mile above ground
Burch.....	31-34	...	62	94-96	Leeds, England

* Kindly collected by L. D. Marinelli of Argonne National Laboratory. From Libby, W. F., Dosages from Natural Radioactivity and Cosmic Rays, *Science*, vol. 122, no. 3158, pp. 57-58, July 8, 1955.

It should be emphasized that the above values of background were in general measured with ionization chambers with appreciable wall thicknesses, say $\frac{1}{8}$ in. of plastic or more. Values obtained with very **thin walled chambers** will be substantially higher. For instance, Sanders (Sanders, A. P., A Radiation Monitor for Measuring A⁴¹ Beta Radiation Dosage at Brookhaven National Laboratory, *Brookhaven Natl. Lab. Rept.* BNL-1304, March, 1953) used a wall thickness of 3.5 mg/cm² and observed an additional soft component at sea level amounting to 0.01 to 0.05 mrad/hr, depending on meteorological conditions.

COSMIC RADIATION

References: Montgomery, "Cosmic Ray Physics," Princeton. Rossi, "High-energy Particles," Prentice-Hall. Wilson, "Progress in Cosmic Ray Physics," North-Holland.

Primary Cosmic Rays. The term "cosmic radiation" is used to designate a complex mixture of naturally occurring high-energy radiations of extraterrestrial origin. The primary cosmic rays are those radiations which arrive from outer space. How-

ever, as soon as these primary rays strike the earth's atmosphere, a variety of secondary radiations are produced. It has been pretty well established that the primary cosmic rays are mostly positively charged particles consisting primarily of protons plus a smaller number of alpha particles. In addition, much smaller numbers of nuclei of many elements in the lower third of the periodic table have been identified.

Peters (Wilson, J. G., "Progress in Cosmic Ray Physics," North-Holland) has pointed out that, although particles heavier than protons compose only 15 per cent of the primary cosmic radiation, they are of considerable importance since they provide 30 to 35 per cent of the incident nucleons, carry up to 30 per cent of the incident cosmic-ray energy, and cause about 50 per cent of the ionization in the upper layers of the atmosphere.

The **energies of the primary cosmic-ray particles** range from a few billion electron volts up to a poorly defined upper limit thought to be of the order of 10^{17} to 10^{19} ev.

Cosmic-ray Secondaries. When the incident cosmic-ray particles collide with the nuclei of elements in the atmosphere, secondary protons, neutrons, and mesons are produced. These in turn produce additional secondaries by additional collisions or by decay. As a result of these processes, the cosmic radiation observed near the earth's surface consists of protons, neutrons, π mesons, μ mesons, electrons, and photons. The process of multiplication of secondaries proceeds through many generations so that as many as 10^8 secondary particles have been known to originate from a single primary. Such events are known as **extensive air showers**. Very few of the primary cosmic rays are able to penetrate to sea level; in fact most of them are absorbed in the upper tenth of the atmosphere.

An absorption curve for cosmic radiation shows that part of the radiation is absorbed quite readily by a few inches of lead while the rest is extremely penetrating. Thus, it is customary to refer to the part that can penetrate an arbitrary thickness of lead, usually 10 cm, as the **"hard" component** while the nonpenetrating portion is referred to as the **"soft" component**. The hard component consists principally of mesons plus very high energy electrons and photons while the soft component is made up of electrons and protons plus a few low-energy mesons, protons, and neutrons.

Intensity of Cosmic Radiation. The intensity of cosmic radiation varies considerably with elevation above sea level and with geomagnetic latitude. The **total intensity** increases as the radiation penetrates the atmosphere, reaching three to five times the original value at a height where pressure is about $\frac{1}{10}$ atm (10 miles). Total intensity decreases steadily below this maximum until at ground level it is only one-fiftieth of the maximum. The ratio between the hard and soft components varies considerably with altitude since the soft component shows a maximum with increasing penetration of the atmosphere whereas the hard component decreases steadily. Thus, the radiation is one-fourth soft and three-fourths hard at sea level whereas, at the 10-mile elevation mentioned above, the soft component is five times as intense as the hard.

The **latitude effect** mentioned above results from the action of the earth's magnetic field in deflecting the incident primary charged particles of the cosmic radiation. At the magnetic poles, particles can get in with far less deflection than at the equator. As a result, cosmic-ray intensity is less for lower geomagnetic latitudes with most of the variation occurring between the equator and a latitude of 45° .

Table 4-3. Estimated Cosmic-ray Intensities at 50° Geomagnetic Latitude *

Altitude		Total intensity		Hard component		Soft component	
Distance, m	Pressure, atm	Omni- directional, particle/ sec-cm ²	Latitude effect, %	Omni- directional, particle/ sec-cm ²	Latitude effect, %	Omni- directional, particle/ sec-cm ²	Latitude effect, %
0	1.000	0.020	10	0.013	10	0.007	10
2,000	0.784	0.035	15	0.018	15	0.017	15
4,500	0.570	0.10	25	0.03	25	0.07	25
10,000	0.261	0.7	45	0.10	30	0.6	30
16,100	0.100	1.5	75	0.25	?	1.25	80
30,000	0.0115	0.5	85	0.4	?	0.06	?
∞	0	0.3	90	?	?	?	?

* From Montgomery, D. J. X., "Cosmic Ray Physics," Princeton, 1949.

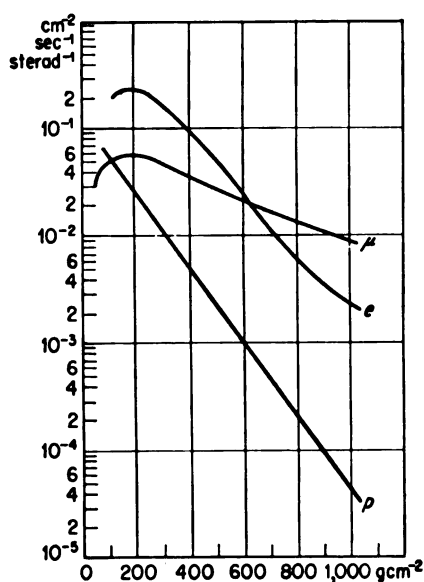


FIG. 4-1. Vertical intensities of various cosmic-ray components at 50° geomagnetic latitude, as functions of atmospheric depth. Curve e : positive and negative electrons of energy greater than 10 Mev. Curve μ : μ mesons of all energies. Curve p : protons of kinetic energy greater than 400 Mev. The last curve represents a crude estimate. (From Rossi, B., "High Energy Particles," Prentice-Hall, 1952.)

The latitude effect amounts to only 10 per cent at sea level but is much greater at higher elevations. In addition to its effect on intensity, the earth's field sets a lower limit on the energy of each type of charged particle below which it cannot reach the earth at a given latitude. For instance, the cutoff energy for vertically directed protons is given by Rossi ("High-energy Particles," Prentice-Hall) as 14 Bev at the equator and 1.5 Bev at 50° geomagnetic latitude.

Table 4-3 and Fig. 4-1 present some representative data illustrating the variations of cosmic-ray intensities with altitude and geomagnetic latitude. A number of interesting facts concerning cosmic rays are shown in Tables 4-4 and 4-5.

Table 4-4. Probable Characteristics of Cosmic Rays Falling upon the Top of the Atmosphere at Various Magnetic Latitudes *

(All energies are given in electron volts)

	Geomagnetic latitude		
	3°	39°	52°
Energy falling per sec on each cm ² of the atmosphere.....	1×10^9	1.7×10^9	3.2×10^9
Total number of ions formed per sec below each cm ² of the upper surface of the atmosphere.....	3×10^7	5.4×10^7	7.4×10^7
Low-energy limit of oncoming particles imposed by the earth's magnetic field.....	15×10^9	8×10^9	2×10^9
Average energy per particle striking the atmosphere.....	3×10^{10}	1.6×10^{10}	0.88×10^{10}
Probable number of particles striking each cm ² of outer surface of the atmosphere per min.....	1.9	6.5	21.8

* From Forsythe, W. E., "Smithsonian Physical Tables," The Smithsonian Institution, 1954.

Table 4-5. Some Cosmic-ray Data *

Total number of rays at top of the atmosphere.	$8 \times 10^{17} \text{ sec}^{-1}$
Total energy carried to earth per second (outer atmosphere).....	$9 \times 10^{18} \text{ Bev/sec}$, $1.4 \times 10^9 \text{ watts}$
If all particles are positively charged this stream gives a current of †.....	0.13 amp
Average number of rays ‡ at top of atmosphere.....	$0.16 \text{ cm}^{-2} \text{ sec}^{-1}$
Average energy of all incident particles, latitude >40°.....	7 Bev
Average energy of all incident particles, all areas, about.....	11 Bev
Cosmic energy reaching earth's outer atmosphere, high latitude.....	$3.8 \times 10^{-3} \text{ erg cm}^{-2} \text{ sec}^{-1}$

Table 4-5. Some Cosmic-ray Data (Continued)

Average energy of the cosmic rays entering the atmosphere is about.....	7×10^9 ev
The spectrum extends from about.....	1×10^9 to 10^{17} ev and probably higher
The energy required for the ionization found in a column 1 cm ² in cross section extending to top of atmosphere at 60°N geomagnetic latitude.....	3.8×10^{-3} erg cm ⁻² sec ⁻¹
Thus in this column there are formed about..	7.4×10^7 ion pairs
This means about.....	90 ion pairs, cm ⁻¹ sec ⁻¹
Total number of rays at sea level from all directions.....	1.2 ray min ⁻¹ cm ⁻²
Cosmic ray at sea level produces ¶.....	1.63 ion pair, cm ⁻³ sec ⁻¹
Total cosmic energy reaching earth per second at sea level.....	40 joules
Radiant-energy flux reaching earth from all stars.....	3.02×10^{-3} erg cm ⁻² sec ⁻¹

* From Forsythe, W. E., "Smithsonian Physical Tables," The Smithsonian Institution, 1954.

† If there were no compensating effects the potential of the earth would increase about 180 volts/sec.

‡ The number varies with the geomagnetic latitude, being about 0.33 particle cm⁻² sec⁻¹ at high latitudes (>40°) and about 0.032 particle cm⁻² sec⁻¹ at the equator. These data are based upon an energy of 32 ev necessary to produce one ion pair.

¶ Thus the average ray entering the cm³ at sea level has an energy of about 10^3 ev.

Importance of Cosmic-ray Studies. Despite the tremendous amount of work already done, cosmic radiation is still the subject of intensive study. It still provides particles of far higher energies than are likely to be created by man and thus will continue to be of interest to physicists as a tool for studying nuclear processes at high energy.

RADIOACTIVITY OF THE EARTH'S CRUST

References: Faul, "Nuclear Geology," Wiley. Kerr, "The Natural Occurrence of Uranium and Thorium," Paper 8/P/1114, International Conference on the Peaceful Uses of Atomic Energy, Geneva, 1955. Rankama, "Isotope Geology," McGraw-Hill. Kohman and Saito, *Ann. Rev. Nuclear Sci.*, vol. 4, 1954.

The radioactivity of the earth's crust, and indeed of all solid substances, is due primarily to the presence of the radioactive substances of the **uranium** and **thorium** series and in some cases of **C-14** and **K-40**. The average concentrations of uranium, thorium, and radium are about 6, 12, and 2×10^{-6} ppm by weight, respectively. When reduced to activity, the average concentrations of uranium, thorium, and radium are all about the same, i.e., 2, 1.3, and 2×10^{-12} curie/g, respectively. These are small concentrations, but it is interesting to note that surface soil having somewhat less than half these average concentrations will contain, in a layer 1 ft thick and 1 sq mile in area, roughly 3 tons of uranium, 6 tons of thorium, and 1 g

of radium. Radon and thoron gases produced by radium and thorium are trapped in rocks and soil, but as noted below, they are continually escaping into the air where they are responsible for a major portion of the atmospheric radioactivity.

The concentrations of radioactive materials found in rocks vary widely depending on location. Some typical values are given in Tables 4-6 to 4-10.

Table 4-6. Mean Radium and Thorium Contents of Various Rock Types in Different Geographic Regions *

(See also Natural Radioactivity in Sec. 6)

Rock type	Ra(10^{-12} g/g)	Th(10^{-6} g/g)
Granites:		
North America, Greenland, Iceland, Ireland, and Japan . .	1.59 ± 0.12	8.1
Finland	4.66 ± 0.40	28.0 ± 2.4
Alps	4.43 ± 0.68	33.0 ± 5.0
South Africa	2.36 ± 0.16	
Basalts:		
North America,† Greenland, Scotland, and Ireland	0.96 ± 0.06	9.8 ± 0.8
England, Germany, France, and Hungary	1.30 ± 0.13	8.8 ± 1.0
Plateau basalts	0.73 ± 0.03	5.2 ± 0.2
Oceanic island basalts	0.90 ± 0.03	4.6 ± 0.3
Dunites (world)	0.42 ± 0.06	3.3 ± 0.3

* From Faul, H., "Nuclear Geology," Wiley, 1954, based on Jeffreys, H., "The Earth," 3d ed., Macmillan, Cambridge, 1952.

† Jeffreys notes that, in view of the variability found by Evans and Goodman (1941), North American basalts are probably too complex to be included under one heading.

Table 4-7. Average Uranium Concentration in Various Rocks

Rock type	U concentration (10^{-6} g/g)
Acid igneous *	3.0
Intermediate igneous *	1.5
Basic igneous *	0.6
Ultrabasic igneous *	0.03
Meteorites *	0.003
Phosphate rock (Fla.) †	120
Bituminous shale (Tenn.) †	50-80
Normal granite †	4

* From Cooper, R. I. B., The Distribution of Radioactivity, *Nature*, vol. 169, p. 350, 1952.

† From Kerr, P. F., "The Natural Occurrence of Uranium and Thorium," Paper P/1114, International Conference on the Peaceful Uses of Atomic Energy, Geneva, 1955.

Table 4-8. Abundance of Potassium and Rubidium in Rocks *

Rock type	% K	% Rb
Granite.....	3.55	0.052
Basalt, diabase.....	0.9	0.01
Crustal rocks, average...	2.7	0.035
Ultramafic rocks.....	0.001	<0.0002
Chondrites.....	0.09	0.0009
Tektites.....	1.77	0.018

* From Kohman, T. P., and N. Saito, *Radioactivity in Geology and Cosmology*, *Ann. Rev. Nuclear Sci.*, vol. 4, p. 401, 1954.

Table 4-9. Average Content of Radioactive Elements and of Their Active Isotopes in Igneous Rocks *

Element	Abundance, ppm	Radioisotope	Abundance, ppm
K	25,900	K ⁴⁰	3.08
Rb	350	Rb ⁸⁷	97.48
In	0.11	In ¹¹⁵	0.11
La	18.23	La ¹³⁸	0.02
Nd	23.9	Nd ¹⁵⁰ (?)	1.34
Sm	6.47	Sm ¹⁴⁷	0.98
Lu	0.75	Lu ¹⁷⁶	0.02
Re	0.05	Re ¹⁸⁷	0.03
Th	11.5	Th ²³²	11.5
U	4	U ²³⁴	0.0002
		U ²³⁵	0.03
		U ²³⁸	3.97

* From Rankama, K., "Isotope Geology," McGraw-Hill, 1955.

Table 4-10. Abundance of Radium in Ocean Sediments and Continental Rocks *

Continental rocks	Radium (10 ⁻¹² g/g)	Ocean sediment	Radium (10 ⁻¹² g/g)
Granite.....	0.2 -5.0	Red clay	3-22
Basalt.....	0.1 -1.0	Globigerina ooze	3-7
Sedimentary..	0.05-0.5	Blue mud	1-3

* From Love, S. K., *Natural Radioactivity of Water*, *Ind. Eng. Chem.*, vol. 43, p. 1541, 1951.

NATURAL RADIOACTIVE BACKGROUND

Table 4-11. Radioactivity of Ocean Water *

Element	Ppm by weight	Ref.
K	380	1
Rb	0.2	1
U	$6.2-28.2 \times 10^{-4}$	2
	13×10^{-4}	3
	20×10^{-4}	4
Th	$1.2-10 \times 10^{-6}$	3
	5×10^{-4}	4
Ra	$0.07-0.58 \times 10^{-10}$	2
	0.7×10^{-10}	3

* Table compiled by W. Lowder, New York Health and Safety Laboratory, AEC. See Background Radiation—A Literature Search, *AEC Rept.* NYO-4712, July, 1956.

1. Sverdrup, H. U., M. W. Johnson, and R. H. Fleming, "The Oceans," Prentice-Hall, 1946.

2. Kohman, T. P., and N. Saito, Radioactivity in Geology and Cosmology, *Ann. Rev. Nuclear Sci.*, vol. 4, p. 401, 1954.

3. Petterson, H., Radium and the Deep Sea, *Am. Scientist*, vol. 41, p. 245, 1953.

4. Legendre, R., *Scientia (Milan)*, vol. 86, no. 6, p. 221, 1951.

Table 4-12. Radioactivity of Natural Waters and Springs *

Element	Water source	Location	Concentration, g/liter	Ref.
Ra	Curie spring	Boulder, Colo.	$267,000 \times 10^{-12}$	1
	Hot spring	Shimane, Japan	$709,800 \times 10^{-12}$	1
	Public water supply	Frankfurt, Germany	$3,000-5,000 \times 10^{-12}$	2
	Public water supply	41 U.S. cities	$0-6.5 \times 10^{-12}$	2
	Normal surface water	U.S. (15 samples)	$0.36-3.41 \times 10^{-12}$	1
	Normal ground water	U.S. (25 samples)	$0.58-3.90 \times 10^{-12}$	1
	River waters	North America	0.03×10^{-12}	3
	Rivers	(Average)	0.07×10^{-12}	4
U	Mineral springs	Japan	$0.002-0.95 \times 10^{-6}$	3
	Great Salt Lake	Utah	5×10^{-6}	3
	River waters	North America	$0.016-0.040 \times 10^{-6}$	3
	Rivers	(Average)	1.0×10^{-6}	4
Th	3 mineral springs	Japan	$2.3-5.0 \times 10^{-6}$	5
	Rivers	(Average)	2×10^{-6}	4

* Table compiled by W. Lowder, Health and Safety Laboratory, AEC.

1. Love, S. K., Natural Radioactivity of Water, *Ind. Eng. Chem.*, vol. 43, p. 1541, 1951.

2. Hursh, J. B., The Radium Content of Public Water Supplies, unclassified report UR-257, May, 1953.

3. Kohman, T. P., and N. Saito, Radioactivity in Geology and Cosmology, *Ann. Rev. Nuclear Sci.*, vol. 4, p. 401, 1954.

4. Koczy, F. F., in Faul, H. (ed.), "Nuclear Geology," Wiley, 1954.

5. Shimokata, K., *J. Chem. Soc. Japan*, vol. 63, p. 1109, 1942.

RADIOACTIVITY OF WATER

Radioactive materials are present almost universally in water. Even rain water picks up radon and thoron, and their active deposits in falling through the air and ground water may become quite radioactive from contact with radioactive substances in the rocks or soil. Ocean water contains small concentrations of radioactive materials, some representative values of which are shown in Table 4-11. Radium is the chief radioactive constituent of natural surface waters. Some data on radioactivity of natural waters and springs are shown in Table 4-12.

The radium content of the water supplies of many United States cities has recently been determined by Hursh. The results given in Table 4-13 show that rather wide variations occur, with values for raw water varying from 0.02×10^{-16} to 65.4×10^{-16} grams of radium per milliliter of water.

Table 4-13. The Radium Content of Public Water Supplies *

City supply	Source	Ra concentration (10^{-16} g Ra/ml H ₂ O)	
		Raw water	Tap water
Atlanta, Ga.....	Chattahoochee R.	0.17	0.09
Baltimore, Md.....	Gunpowder R.	0.20	0.08
Birmingham, Ala.....	Cahaba R., L. Purdy	0.24	0.17
Bismarck, N.Dak.....	Missouri R.	2.43	0.26
Boise, Idaho.....	{ Shallow wells 75% } { Deep wells 25% }	1.03	0.96
Boston, Mass.....	Nashua R.	0.14	0.17
Buffalo, N.Y.....	Lake Erie	0.35	0.28
Charleston, S.C.....	Edisto R.	1.81	1.41
Charleston, W.Va.....	Elk R.	0.41	0.45
Cheyenne, Wyo.....	Surface and wells	0.50	0.34
Chicago, Ill.....	L. Michigan	0.24	0.29
Cincinnati, Ohio.....	Ohio R.	0.61	0.33
Cleveland, Ohio.....	L. Erie	0.33	0.23
Dallas, Tex.....	Garza and Bachman Lakes	0.85	0.32
Denver, Colo.....	South Platte R.	0.77	0.47
Detroit, Mich.....	Detroit R.	0.26	0.18
Indianapolis, Ind.....	Fall Cr. and White R.	1.37	0.86
Joliet, Ill.....	Deep wells	65.4	57.9
LaVerne, Calif.....	Colorado R.	1.00	0.35
Los Angeles, Calif.....	Owens Valley Aqu.	0.09	0.08
Louisville, Ky.....	Ohio R.	0.84	0.41
Miami, Fla.....	Shallow wells	4.78	1.68
Memphis, Tenn.....	Shallow and deep wells	2.11	1.70
New Orleans, La.....	Mississippi R.	4.25	0.16

* From Hursh, J. B., The Radium Content of Public Water Supplies, *AEC Rept.* UR-257, May, 1953.

Table 4-13. The Radium Content of Public Water Supplies (Continued)

City supply	Source	Ra concentration (10^{-18} g Ra/ml H_2O)	
		Raw water	Tap water
New York, N.Y.....	Catskill supply	0.15	0.18
New York, N.Y.....	Croton supply	0.20	0.18
Oklahoma City, Okla....	N. Canadian R.	1.06	0.42
Omaha, Nebr.....	Missouri R.	17.7	0.54
Philadelphia, Pa.....	Delaware R.	0.48	0.44
Phoenix, Ariz.....	Along Verdi R.	0.27	0.18
Pierre, S.Dak.....	Shallow wells along Missouri R.	0.62	0.15
Pittsburgh, Pa.....	Allegheny R.	37.	1.41
Portland, Ore.....	Bull Run R.	0.14	0.01
Raleigh, N.C.....	Walnut Cr.	0.22	0.25
Richmond, Va.....	James R.	0.33	0.23
Sacramento, Calif.....	Sacramento R.	0.18	0.15
Salt Lake City, Utah....	Cottonwood Cr.	0.34	0.50
San Francisco, Calif....	Calaveras Res.	0.18	0.07
St. Louis, Mo.....	Mississippi R.	10.8	0.28
Tacoma, Wash.....	Green R.	0.02	0.00
Washington, D.C.....	Potomac R.	0.33	0.27
Wichita, Kans.....	Deep wells	2.27	0.75

RADIOACTIVITY OF AIR

The radioactivity of air is due principally to the **radon** and **thoron** that diffuse from the ground. The active decay products of these gases are solid substances and become attached readily to dust particles in the air. Such dust is picked up by rainfall which will cause some increase in the background counting rate of an exposed Geiger counter as shown in Fig. 4-2. Figure 4-3 is a plot of the activity of a continuous dust monitor compared with the temperature difference between points 410 and 18 ft above the ground. It is seen that sizable build-up of activity occurs during periods of temperature inversion when vertical mixing by thermal currents does not occur. Air activity is also influenced by other factors that affect the rate of release of radon and thoron from the ground, e.g., a snow cover, and by the amount of dust and smoke in the air (Fig. 4-2 and Fig. 4-3).

The actual concentrations of radon noted by different observers vary widely. Eisenbud and Harley (*Science*, vol. 117, p. 141, Feb. 13, 1953) give 5×10^{-14} curie/liter as a conservative estimate for normal air but point out that values may frequently be as high as 5×10^{-13} curie/liter. Anderson, Mayneord, and Turner found an average concentration of 2 to 3×10^{-12} curie/liter on a London rooftop. Radon concentrations in metropolitan New York City have been studied by Glauberman and Breslin (Glauberman, H., and A. J. Breslin, Environmental Radon Concentrations, An Interim Report, *AEC Rept.* NYO-4861). Values in **uranium** or **radium** mines can be very high (10^{-9} curie/liter or more) depending on the condi-

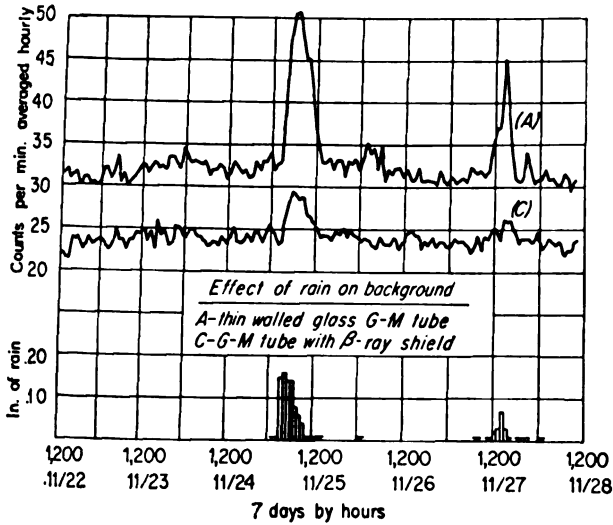


Fig. 4-2. Effect of rain on background counting rate at Upton, N.Y., 1949.

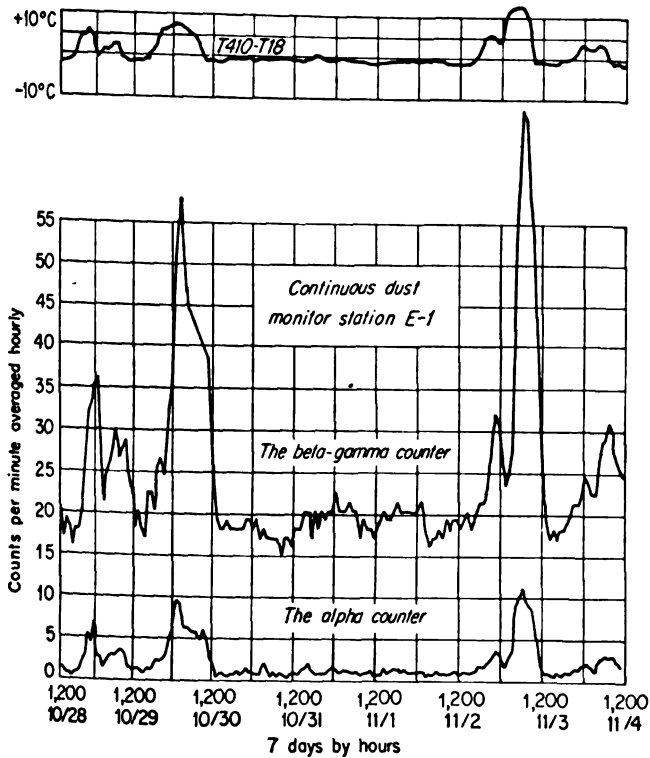


Fig. 4-3. Activity of atmospheric dust at Upton, N.Y., 1949.

tions of ventilation. In fact, mines and other underground spaces in general appear to have fairly high air concentrations of radon even in areas where there are no deposits of commercially important uranium. Table 4-14 gives some values of this

Table 4-14. Radon Concentrations in Mines in New York State *

Area	Type	Concentrations in 10^{-11} curie/liter		
		Conversion †	Calculated ‡	Flask ¶
A	Cavern	2.0-15.0	1.4-11.1	1.2-11.0
B	Iron mine	6.1-14.2	4.7-10.6	3.3-4.1
C	Iron mine	6.9-11.2	5.3-8.5	3.2-3.7
D	Talc mine	1.6-6.2	1.2-4.6	2.0
E	Hematite mine	2.1-5.2	1.6-4.0	Lost in counting
F	Talc mine	1.5-1.7	1.0-1.3	0.8
G	Iron mine	0.7-1.7	0.5-1.3	0.2-1.2
H	Talc mine	0.6-1.6	0.5-1.2	0.8
I	Zinc mine	0.2-0.6	0.1-0.4	0.01-0.1
J	Zinc mine	0.1-0.3	0.1-0.2	0.01-0.1
K	Zinc mine	0.28	0.21	0.21
L	Gypsum mine	No noticeable reading		0.13
M	Rock-salt mine	No noticeable reading		0.03
N	Abandoned mine	No noticeable reading		0.01

* From Harris, S. J., Radon Levels Found in Mines in New York State, *N.Y. State Dept. Labor, Monthly Rev.*, vol. 33, p. 10, October, 1954.

† These figures converted directly from disintegration per minute per liter, on filter-paper samples of dust.

‡ This column calculated from same data as the conversion figures using a formula developed by Dr. J. Harley (NYOO, AEC).

¶ These figures obtained from direct sampling for radon, using 1-liter glass flasks and counting electronically.

kind obtained by Harris in New York State. It is of interest to note that lung exposure is appreciable for levels such as are shown in this table. Harris (Harris, S. J., Radon Levels Found in Mines in New York State, *N.Y. State Dept. Labor, Monthly Rev.*, vol. 33, p. 10, October, 1954) quotes a value, furnished by Bale, of 22.6 rems per normal work week at a level of 10^{-10} curie/liter.

Small amounts of C-14 and H-3 are found in the atmosphere. They are produced in the upper layers by the action of cosmic rays (Libby, *Phys. Rev.*, vol. 69, p. 671, 1946). However, their half-lives are long enough so that they are thoroughly mixed with the atmosphere as a whole and become incorporated in animals and plants to the extent of 0.95×10^{-10} per cent (Anderson *et al.*, *Phys. Rev.*, vol. 72, p. 931, 1947). Since no more C-14 will be taken up after a plant or animal dies, estimates of the age of fossil deposits may be based on their C-14 content. Values of tritium abundance given by Kaufman and Libby (*Phys. Rev.*, vol. 93, p. 1337, Mar. 15, 1954) range from 0.5 to 67 tritium atoms per 10^{18} hydrogen atoms. These authors estimate that the earth carries a total tritium inventory of 1,800 g with only about 1 per cent in the atmosphere.

RADIOACTIVITY OF THE HUMAN BODY

Since radioactive materials are almost universally distributed at low concentrations in vegetable and mineral matter, it is inevitable that measurable quantities should be found in the human body. K-40, C-14, and Ra-226 plus its decay products are the most important isotopes in this connection. In recent years, some additional radioactive elements originating in tests of nuclear weapons have been noted. (Anderson, E. C., *et al.*, Radioactivity of People and Foods, *Science*, vol. 125, p. 1273, June 28, 1957; U.S. Congress Joint Committee on Atomic Energy, "The Nature of Radioactive Fallout and its Effects on Man," U.S. Government Printing Office, 1957; Health and Safety Laboratory, AEC, Environmental Contamination from Weapons Tests, *AEC Rept. HASL-42*.)

Potassium is the most abundant radioactive material in the human body. Estimates range from 100 to 300 g for a 70-kg man. A number of values are shown in Table 4-15, the weighted average for these 268 individuals amounting to 0.17 per cent or 120 g per 70-kg "standard man." Potassium is concentrated in muscle tissue with relatively little in bone or fat.

Table 4-15. Potassium Content in Living Subjects *

Author	Subjects	Age	Avg weight, kg	% K
Corsa (1950).....	30 M	21-32	...	$0.18_1 \pm 0.01_9$
Aikawa (1952).....	20 F		...	$0.12_3 \pm 0.01_1$
Edelman (1952).....	14 F		...	0.16
	33 M		...	0.18
Blainey (1954).....	17 M		...	$0.17_9 \pm 0.01_4$
	7 F		...	$0.13_7 \pm 0.01_9$
Rundo and Sagild (1955).....	6 M	22-31	76	$0.19_0 \pm 0.02$
	4 F	17-28	61	$0.16_7 \pm 0.01_7$
Burch and Spiers (1954).....	10 M + 3 F	19-20	...	$0.21_5 \pm 0.01$
	11 M	26-41	...	$0.21_2 \pm 0.01$
	4 M	60-79	...	$0.21_5 \pm 0.02$
Rundo and Sagild (1955).....	6 M	22-31	76	$0.21_5 \pm 0.01_2$
	4 F	17-28	61	$0.19_7 \pm 0.01_5$
Sievert (1955).....	20 M	10-13	40	$0.19_1 \pm 0.008_3$
	11 M	20-29	70	$0.19_6 \pm 0.01_0$
	10 M	30-49	70	$0.20_9 \pm 0.01_3$
	14 M	62-84	70	$0.15_5 \pm 0.008_2$
	17 F	10-13	40	$0.16_4 \pm 0.01_1$
	4 F	25-28	70	$0.14_5 \pm 0.005_2$
	5 F	36-56	70	$0.13_8 \pm 0.009_3$
	18 F	66-86	70	$0.12_9 \pm 0.008_6$

* From Sievert, R. M., "Measurements of Low-level Radioactivity, Particularly the γ -radiation from Living Subjects," Paper 8/P/792, International Conference on the Peaceful Uses of Atomic Energy, Geneva, 1955.

Table 4-16. Mean Values of Body Radium for Groups of Subjects *

Group	No. of subjects	Years		Units of 10^{-10} g Ra	
		Mean age	Mean time at Stateville	Mean body content	Standard deviation of mean
Chicago adult.....	1	29	0.00	0.40	
Stateville:					
New.....	11	27	0.32	1.00	0.14
Intermediate.....	8	38	7.6	2.02	0.28
Long term.....	11	44	19.7	2.36	0.30
Chicago boys.....	7	16.6		0.36	0.14
Lockport boys.....	8	16.6		3.68	0.50

* From Stehney, A. F., and H. F. Lucas, Jr., "Studies on the Radium Content of Humans Arising from the Natural Radium of Their Environment," Paper 8/P/852, International Conference on the Peaceful Uses of Atomic Energy, Geneva, 1955.

Table 4-17. Exposure of Man to Natural Background Radiation *

Type of exposure	Exposure, mreps/day	Exposure, mrems/day	Remarks
External exposure to U, Th, and Ac isotope series	0.13	0.13	Attributable mostly to Rn and Tn products in air and on earth's surface. Frequently less by factor of 5 when no atmosphere inversion exists
External exposure to cosmic rays at sea level and 50° geomagnetic latitude	0.07	0.07	Increases 75% in going to 2,000 meters elevation. Decreases 10% in going to 0° geomagnetic latitude
Internal exposure to Ra and its products	0.013	0.25	Assumes 1.18×10^{-10} g Ra in body. This may vary by factor of 5 either way
Internal exposure to K^{40} ..	0.15	0.15	Assumes 245 g K or 0.18 microcurie K^{40} in body, 75% of which is in muscle
Total exposure.....	0.36	0.60	It is not uncommon for the total exposure to vary from these values by $\pm 50\%$

* From Morgan, K. Z., *AEC Rept. TID-5031*.

Considerable attention has been focused on the radium content of the human body because of the extent to which this material has been used commercially and in hospitals. It has been found that the body content depends on the radium content of both the food and water consumed, the values being of the order of 10^{-10} g of radium per individual. Table 4-16 illustrates this point. It summarizes the results of studies made at Stateville and Lockport, Ill., where the drinking water contains 3.4×10^{-12} g/liter and 8×10^{-12} g/liter of radium, respectively, compared with similar measurements on Chicago residents where the drinking water contains only 0.03×10^{-12} g/liter of radium.

The total radiation dosage received by human beings will be partly due to external radiation and partly due to internal emitters. Estimates of total dosage and of its various components published by two authors are given in Tables 4-17 and 4-18.

Table 4-18. Radiation Dosages of the Human Body Due to the Natural Radiation Background *

<i>Source of dose</i>	<i>Dosage, mr/year</i>
Potassium in the body.....	19
Carbon in the body.....	1.5
Radium (bones only, uniform distribution).....	6.7
Radium (bones only, nonuniform distribution) ..	67†
Cosmic rays at sea level.....	33-37
Cosmic rays at 5,000 ft elevation.....	40-60
Cosmic rays at 10,000 ft elevation.....	80-120
Cosmic rays at 15,000 ft elevation.....	160-240
Cosmic rays at 20,000 ft elevation.....	300-450

* From Libby, W. F., Dosages from Natural Radioactivity and Cosmic Rays, *Science*, vol. 122, no. 3158, p. 57, July 8, 1955.

† The radium content of the human body is based on data of A. F. Stehney of Argonne National Laboratory.

General References

- Libby, W. F., Dosages from Natural Radioactivity and Cosmic Rays, *Science*, vol. 122, no. 3158, pp. 57-58, July 8, 1955.
- Montgomery, "Cosmic Ray Physics," Princeton.
- Rossi, "High-energy Particles," Prentice-Hall.
- Wilson, "Progress in Cosmic Ray Physics," North-Holland.
- Forsythe, W. E., "Smithsonian Physical Tables," The Smithsonian Institution, 1954.
- Faul, H., "Nuclear Geology," Wiley, 1954, based on Jeffreys, H., "The Earth," 3d ed., Macmillan, Cambridge, 1952.
- Cooper, R. I. B., The Distribution of Radioactivity, *Nature*, vol. 169, p. 350, 1952.
- Kerr, P. F., "The Natural Occurrence of Uranium and Thorium," Paper P/1114, International Conference on the Peaceful Uses of Atomic Energy, Geneva, 1955.
- Kohman, T. P., and N. Saito, Radioactivity in Geology and Cosmology, *Ann. Rev. Nuclear Sci.*, vol. 4, p. 401, 1954.
- Rankama, K., "Isotope Geology," McGraw-Hill, 1955.
- Love, S. K., Natural Radioactivity of Water, *Ind. Eng. Chem.*, vol. 43, p. 1541, 1951.

Sverdrup, H. U., M. W. Johnson, and R. H. Fleming, "The Oceans," Prentice-Hall, 1946.

Petterson, H., Radium and the Deep Sea, *Am. Scientist*, vol. 41, p. 245, 1953.

Legendre, R., *Scientia (Milan)*, vol. 86, no. 6, p. 221, 1951.

Koczy, F. F., in Faul, H. (ed.), "Nuclear Geology," Wiley, 1954.

Shimokata, K., *J. Chem. Soc. Japan*, vol. 63, p. 1109, 1942.

Hursh, J. B., The Radium Content of Public Water Supplies, *AEC Rept.* UR-257, May, 1953.

Harris, S. J., Radon Levels Found in Mines in New York, *N.Y. State Dept. Labor, Monthly Rev.*, vol. 33, p. 10, October, 1954.

Sievert, R. M., "Measurements of Low-Level Radioactivity, Particularly the γ -radiation from Living Subjects," Paper 8/P/792, International Conference on the Peaceful Uses of Atomic Energy, Geneva, 1955.

Stehney, A. F., and H. F. Lucas, Jr., "Studies on the Radium Content of Humans Arising from the Natural Radium of Their Environment," Paper 8/P/852, International Conference on the Peaceful Uses of Atomic Energy, Geneva, 1955.

Morgan, K. Z., *AEC Rept.* TID-5031.

Sedlet, J., Environmental Radioactivity at Argonne National Laboratory—Report for the Year 1955, *Rept.* ANL-5684, February, 1957.

Blifford, I. H., *et al.*, Radioactivity of the Air, *Naval Research Lab. Rept.* NRL-4760, June, 1956.

Background Radiation—A Literature Search, *AEC Rept.* NYO-4712, July, 1956.

Glauberger, H., and A. J. Breslin, Environmental Radon Concentrations, An Interim Report, *AEC Rept.* NYO-4861.

Anderson, E. C., *et al.*, Radioactivity of People and Foods, *Science*, vol. 125, p. 1273, June 28, 1957.

U.S. Congress Joint Committee on Atomic Energy, "The Nature of Radioactive Fallout and its Effects on Man," U.S. Government Printing Office, 1957.

Health and Safety Laboratory, AEC, Environmental Contamination from Weapons Tests, *AEC Rept.* HASL-42.

Section 5

IONIZING RADIATION

By

DAVID HALLIDAY, *University of Pittsburgh, Department of Physics.*

IONIZING RADIATION

David Halliday

General References

Rose, M. E., *et al.*, in Seigbahn, K. (ed.), "Beta and Gamma Ray Spectroscopy," Interscience, 1955.

Feingold, A. M., *Revs. Mod. Phys.*, vol. 23, p. 10, 1951.

Baggerly, L. J., P. Marmier, F. Boehm, and J. W. M. DuMond, *Phys. Rev.*, vol. 100, p. 1364, 1955.

Gell-Mann and Rosenblum, *Sci. American*, July, 1957.

Perlman and Ypsilantis, *Phys. Rev.*, vol. 79, p. 30, 1950.

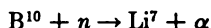
Perlman, Ghiorso, and Seaborg, *Phys. Rev.*, vol. 77, p. 26, 1950.

Albert and Wu, *Phys. Rev.*, vol. 74, p. 847, 1948.

Blatt and Weisskopf, "Theoretical Nuclear Physics," Wiley, 1952.

Goldhaber and Sunyar, *Phys. Rev.*, vol. 83, p. 906, 1951.

Ionizing radiations are of two general classes. Ions are produced in solids, liquids, or gases when **charged particles** pass through them. Ions can also be produced indirectly by **uncharged particles**, such as neutrons or gamma rays. A neutron, for example, can transfer energy to a proton in a collision; the proton then produces ionization. Also, a neutron may be absorbed by a nucleus, such as B-10, resulting in the appearance of an alpha particle according to the reaction



This alpha particle, and also the recoiling Li-7 nucleus, then produce ions as they move through the absorbing medium.

Gamma rays can also produce secondary charged particles (electrons) by three main processes (see below).

Table 5-1 shows the particles now known to modern physics along with some of their characteristics. Many of these particles are produced in the cosmic radiation or in the large accelerators described in Sec. 6. These particles are produced only at very high energies and will not come to the consideration of many radiation physicists.

A **radioactive nucleus** or **radionuclide** is one that changes spontaneously to a lower-energy state, emitting a particle or particles in the process. Particles commonly emitted are alpha particles, electrons, positrons, and gamma rays.

The law governing the decay of a single radionuclide can be derived from the following assumption: the **rate of decay of radioactive atoms** in a sample is propor-

tional to the number of such atoms in that sample. The law that follows by simple integration from this statement is

$$N = N_0 e^{-\lambda t} \quad (5-1)$$

N is the number of atoms existing at any time t ; N_0 is the number present at zero time. λ is called the **decay constant** for the transition.

Table 5-1. Particles of Modern Physics as of July, 1957 *

Particle	Spin	Rest mass (electron masses)	Mean life, sec	Decay products		
Xi	Ξ^-	$\frac{1}{2}$	2,585	10^{-10} - 10^{-9}	$\Lambda^0 + \pi^-$	
	Ξ^0	$\frac{1}{2}$	Not yet found			
Sigma	Σ^+	$\frac{1}{2}$	2,325	0.7×10^{-10}	$p + \pi^0; n + \pi^+$	
	Σ^-	$\frac{1}{2}$	2,341	1.5×10^{-10}	$n + \pi^-$	
	Σ^0	$\frac{1}{2}$	2,324	Not measured	$\Lambda^0 + \gamma$	
Lambda	Λ^0	$\frac{1}{2}$	2,182	2.7×10^{-10}	$p + \pi^-; n + \pi^0$	
Proton	p	$\frac{1}{2}$	1,836.1	Stable		
Neutron	n	$\frac{1}{2}$	1,838.6	About 1,000	$p + e^- + \bar{\nu}$	
K meson	K^+	0	966.5	1.2×10^{-8}	$\mu^+ + \nu; \pi^+ + \pi^0$ $\pi^+ + \pi^+ + \pi^-$ $\pi^+ + \pi^0 + \pi^0$ $\mu^+ + \nu + \pi^0$ $e^+ + \nu + \pi^0$	
	K^-	0	966.5	1.2×10^{-8}	$\mu^- + \bar{\nu}; \pi^- + \pi^0$ $\pi^- + \pi^- + \pi^+$ $\pi^- + \pi^0 + \pi^0$ $\mu^- + \bar{\nu} + \pi^0$ $e^- + \bar{\nu} + \pi^0$	
	K_1^0	0	965	1×10^{-10}	$\pi^+ + \pi^-; \pi^0 + \pi^0$	
	K_2^0	0	965	3×10^{-8} - 10^{-6}	$\pi^+ + e^- + \bar{\nu}; \pi^- + e^+ + \nu$ $\pi^+ + \mu^- + \bar{\nu}$ $\pi^- + \mu^+ + \nu$ $\pi^+ + \pi^- + \pi^0$ $\pi^0 + \pi^0 + \pi^0$	
					$\mu^+ + \nu$ $\mu^- + \bar{\nu}$ $\gamma + \gamma$	
	Pion	π^+	0	273.2	2.6×10^{-8}	
		π^-	0	273.2	2.6×10^{-8}	
		π^0	0	264.2	10^{-16} - 10^{-15}	
	Muon	μ^-	$\frac{1}{2}$	206.7	2.2×10^{-6}	$e^- + \nu + \bar{\nu}$
	Electron	e^-	$\frac{1}{2}$	1	Stable	
Neutrino	ν	$\frac{1}{2}$	0	Stable		
Photon	γ	1	0	Stable		

* From Gell-Mann and Rosenblum, *Sci. American*, July, 1957.

We often describe a radioactive transition by giving its **half-life**. This is the time after which any given number of radioactive atoms will be reduced to half. It can be shown from Eq. (5-1) that the half-life is related to the disintegration constant by

$$t_{1/2} = \frac{0.693}{\lambda} \quad (5-2)$$

The **mean life** $\bar{t}$ is also often used. As its name suggests, it is simply the average lifetime for all the atoms in a given radioactive sample. It is related to the disintegration constant by

$$\bar{t} = \frac{1}{\lambda} \quad (5-3)$$

Since the rate of decay of a radioactive sample is proportional to the number of atoms present, we can write from Eq. (5-1)

$$R = R_0 e^{-\lambda t} \quad (5-4)$$

where R is the **decay rate** at any time t and R_0 is the rate at zero time. This equation holds true whether R and R_0 are absolute counting rates or, as is usually the case, they are simply the readings on a detector placed near the sample.

In practice, **half-lives** are measured by plotting the reading of a suitable detector (corrected for background effects and counting losses) against time on semilogarithm paper as in Fig. 5-1. If the decay is a simple one, a straight line results. The half-life can easily be found from its slope.

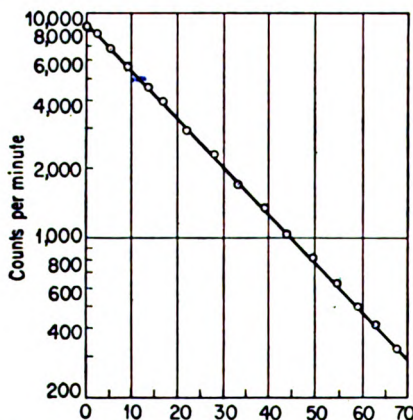


FIG. 5-1. Decay of P-32 (half-life = 14.3 days). Readings are corrected for instrumental background and counting losses.

Sometimes, one deals with a **mixture of radionuclides**, each decaying with its own half-life independently of the others. Such mixtures may also be analyzed by plotting the counting rate as a function of time on semilogarithm paper as in Fig. 5-2. The counting is continued for such a time that the slope of the curve becomes constant. When this happens, we can be reasonably sure that we are now dealing with the decay of a single radionuclide and that the shorter-lived species have essentially disappeared. This limiting slope gives the half-life of this longest-lived species.

By extending this limiting slope toward zero time and subtracting this straight line from the experimental curve, the decay curve for the rest of the sample is obtained. If it proves to be a straight line (as in Fig. 5-2) we may conclude with reasonable certainty that there were only two radionuclides in the sample. The slope of this second straight line gives us the half-life of this second radionuclide.

If the difference curve in Fig. 5-2 had proved not to be a straight line, it would be necessary to repeat the process until a straight line was obtained. This method of analysis is suited only for cases in which there are a small number of radionuclides with rather well separated half-lives.

Sometimes the product of a radioactive decay is itself radioactive. Such decay chains, extending through as many as a dozen nuclides, are found in nature at the heavy end of the isotope chart and also among fission products in the middle region of this chart.

We first analyze the case in which species 1 is decaying into species 2 which in turn decays into species 3. The number of atoms of type 1 and of type 2 present at any time t in a radioactive sample is given by

$$N_1 = N_{1,0}e^{-\lambda_1 t} \quad (5-5)$$

$$N_2 = N_{2,0}e^{-\lambda_2 t} + \frac{\lambda_1}{\lambda_2 - \lambda_1} N_{1,0}(e^{-\lambda_1 t} - e^{-\lambda_2 t}) \quad (5-6)$$

If no atoms of type 2 are present, initially the number of type 2 nuclei will increase from zero, pass through a maximum, and then decrease, as Fig. 5-3 shows.

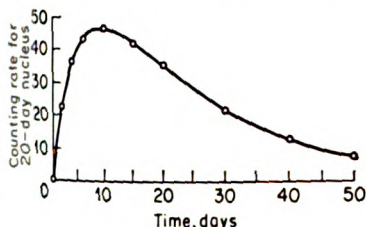


Fig. 5-3. Hypothetical build-up and decay for a nucleus with a mean life of 5 days decaying into one with a mean life of 20 days.

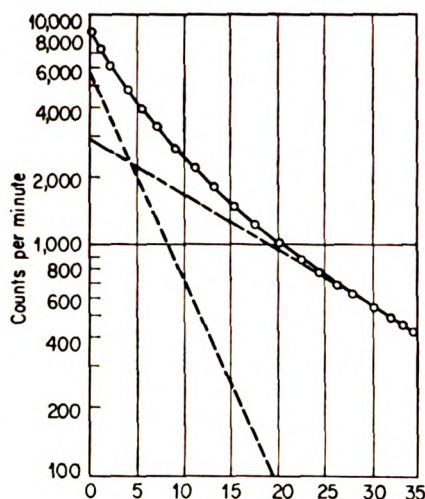


Fig. 5-2. Hypothetical decay curve for a sample containing Cu-64 (12.8 hr) and Cu-61 (3.4 hr).

If λ_2 is very much greater than λ_1 , transient equilibrium can occur. For this condition Eq. (5-6) reduces, after enough time has elapsed, to

$$N_2 = \frac{\lambda_1 N_{1,0}}{\lambda_2} e^{-\lambda_1 t} \quad (5-7)$$

Note that species 2 decays with the half-life characteristic of species 1.

If, in addition to the above condition, we also have the condition that the half-life of species 1 is very much greater

than the length of any experiment that we plan to conduct, we can assume that

the term $e^{-\lambda_1 t}$ in Eq. (5-7) never departs essentially from unity. Equation (5-7) then reduces to

$$N_2 = \frac{\lambda_1}{\lambda_2} N_{1,0} \quad (5-8)$$

Under these conditions the number of atoms of type 2 remains constant. This is called **secular equilibrium**. Radioactive atoms of type 2 are being produced at a certain rate and are decaying at exactly the same rate. Such equilibrium will occur whenever a radionuclide is produced at a constant rate. This may occur by the decay of a long-lived parent or by production at a constant rate in a cyclotron or a reactor.

A general formula for **radioactive chain decay** was given by Bateman in 1910. It is, for the case in which only atoms of type 1 are present initially,

$$N_n = c_1 e^{-\lambda_1 t} + c_2 e^{-\lambda_2 t} + \dots + c_n e^{-\lambda_n t} \quad (5-9)$$

where

$$c_1 = \frac{\lambda_1 \lambda_2 \dots \lambda_{n-1}}{(\lambda_2 - \lambda_1)(\lambda_3 - \lambda_1) \dots (\lambda_n - \lambda_1)} N_{1,0}$$

$$c_2 = \frac{\lambda_1 \lambda_2 \dots \lambda_{n-1}}{(\lambda_1 - \lambda_2)(\lambda_3 - \lambda_2) \dots (\lambda_n - \lambda_2)} N_{1,0}$$

etc.

Equations (5-5) and (5-6) (the latter with $N_{2,0}$ taken as zero) are special cases of Eq. (5-9).

Alpha emission occurs largely but not entirely at the heavy end of an isotope chart. Like fission, it is a symptom of the tendency of heavy nuclei to break up because of coulomb repulsion.

The main measurable characteristics of alpha decay are half-life and disintegration energy. The disintegration energy E is larger than the energy of the alpha particle E_α because it includes the recoil energy of the daughter nucleus. These two quantities are related simply by

$$E = E_\alpha \left(1 + \frac{m}{M} \right) \quad (5-10)$$

where m is the mass of the alpha particle and M that of the daughter nucleus.

It was noted in 1911 by Geiger and Nuttall that there was a relationship between half-life (or disintegration probability) and disintegration energy. In general, the shorter-lived alpha emitters had larger disintegration energies. This relationship is called the **Geiger-Nuttall law**. The dependence of λ on E is extremely sensitive. A factor of 2 in energy corresponds to a factor of about 10^{24} in disintegration probability.

During and after World War II, many new alpha emitters were discovered as a result of nuclear investigations at the heavy end of the isotope chart. It became possible to plot **Geiger-Nuttall curves** for alpha emitters of the same element. Figure 5-4 shows a number of such plots. It is clear that there is considerable regularity and that the relationships are quite smooth.

There is also a relationship between the disintegration constant of an alpha emitter and its position in the isotope chart. Figure 5-5 shows the experimental curves in which $\log \lambda$ is plotted against A . In general, adding neutrons to a nucleus

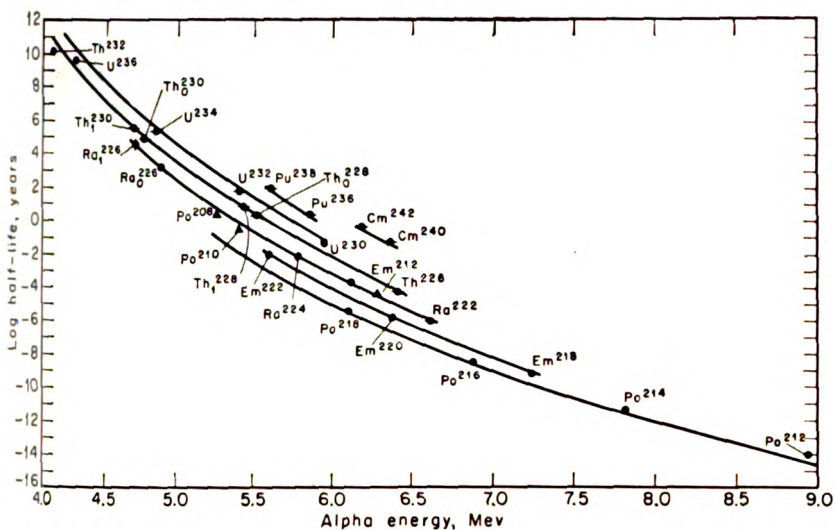


Fig. 5-4. A semilog plot of half-life vs. energy for nuclei with N and Z both even. Isotopes are connected by solid lines. (From Perlman and Ypsilantis, *Phys. Rev.*, vol. 79, p. 30, 1950.)

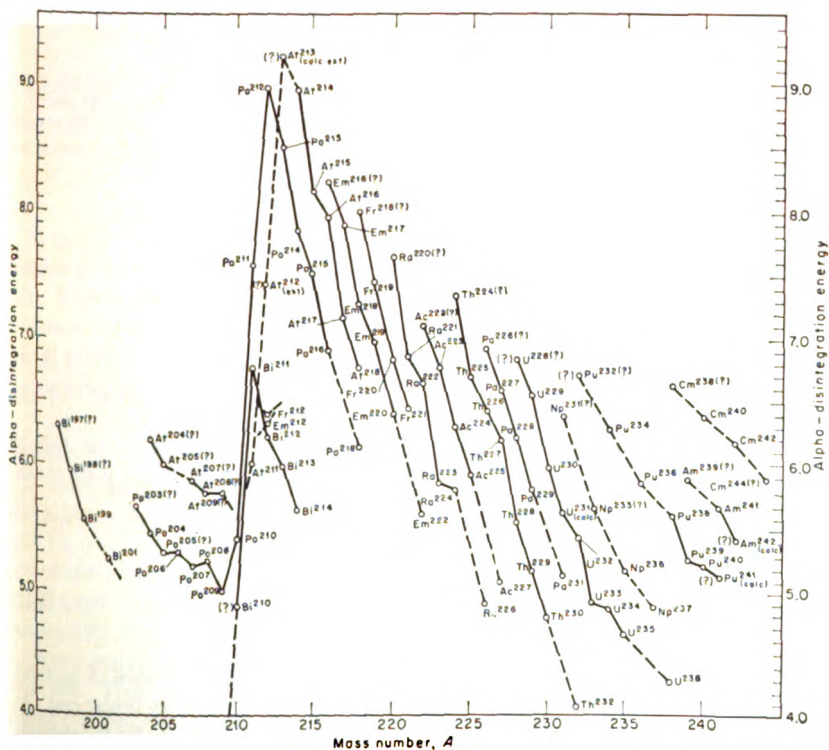


Fig. 5-5. A plot of energy vs. mass number for alpha emitters. Isotopes are connected by lines. (From Perlman, Ghiorso, and Seaborg, *Phys. Rev.*, vol. 77, p. 26, 1950.)

increases its stability against alpha decay. The deep irregularity near $A = 206$ is caused by **closed-shell effects** within the nucleus.

The Geiger-Nuttall relationship can be understood in the terms of the **barrier-tunneling theory of alpha decay**. Figure 5-6 shows a plot of the potential energy of an alpha particle in the field of its residual nucleus. The horizontal line marked E shows the energy of the alpha particle after it has been emitted. One visualizes that the alpha particle exists within the nucleus in a preformed state with this same total energy E . Classically, it could never penetrate the potential barrier that exists between the points marked a and b . According to quantum mechanics, however, there is always a finite chance that such particles will penetrate a classically forbidden barrier.

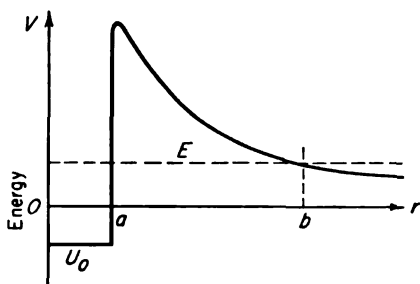


FIG. 5-6. An assumed plot of V , the potential energy of an alpha-particle daughter-nucleus system as a function of r , the distance between centers. The potential energy is taken to be zero when the particles are a large distance apart. E is the disintegration energy.

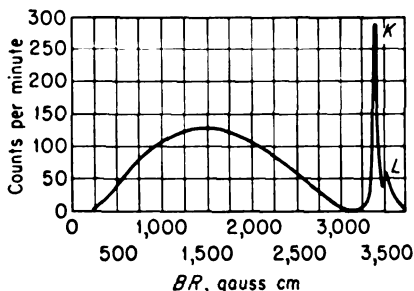


FIG. 5-7. The broad distribution shows electrons emitted from a Cs-137 source, as a function of momentum. The peaks marked K and L are internal conversion peaks; see p. 5-12.

This theory explains at once why nuclei that are unstable against alpha emission do not immediately decay; the barrier penetrability in these cases is small. The main features of the **Geiger-Nuttall law** are also clear. If an alpha particle is emitted with a relatively high energy this means that, other things being equal, the barrier through which it has "leaked" is a narrower one than otherwise. The penetrability factor is larger for such narrow barriers.

This theory of alpha decay was first worked out quantitatively in 1926 by Gamow and by Gurney and Condon. The theory is still being refined at the present time.

By **beta decay**, we understand



or the capture by the nucleus of an electron from its extranuclear shell (K capture). Figure 5-7 shows a **beta spectrum** for the emitted electrons in a particular case. An important feature is the existence of a maximum energy, 0.540 Mev for this example. In early studies of beta decay, the existence of the spectrum was puzzling and seemed to suggest that the law of conservation of energy failed during such processes.

Evidently the disintegrating nucleus proceeds from one well-defined energy state to another similarly well-defined state. However, varying amounts of energy are carried away by the escaping electron.

It is possible to prove that a definite amount of energy Q is released in each beta process and, furthermore, that this energy release is equal to the maximum energy of the beta spectrum. One needs to know the atomic masses of the parent and daughter nuclei. For negative electron emission and also (closely) for K capture, the Q is given by

$$Q = (M_z - M_{z+1})c^2 \quad (5-12)$$

in which the M 's are *atomic* rather than nuclear masses. For positron emission, we have

$$Q = (M_z - M_{z-1})c^2 - 2m_e c^2 \quad (5-13)$$

in which m_e is the electron mass.

Consider the beta emitter N^{13} which decays according to the scheme



Consider also the reaction



Applying the combined conservation of mass and energy law to Eq. (5-15) yields

$$2.98 \text{ Mev} = (N^{13} + n - C^{13} - p)c^2$$

in which the symbols represent atomic masses and in which $+2.98 \text{ Mev}$ is the Q of the reaction of Eq. (5-12). This allows us to write

$$\begin{aligned} N^{13} - C^{13} &= 2.98 \text{ Mev} - (n - p)c^2 \\ &= 2.20 \text{ Mev} \end{aligned}$$

For the particular case of Eq. (5-14) we have, combining Eq. (5-13) with the above result,

$$\begin{aligned} Q &= 2.20 \text{ Mev} - 1.02 \text{ Mev} \\ &= 1.18 \text{ Mev} \end{aligned}$$

Within experimental error, this energy release equals the maximum energy of the beta spectrum.

The difficulty with the conservation of energy law, as well as equally serious difficulties with the laws of conservation of angular momentum and of statistics, was cleared up by Pauli with the suggestion of the **neutrino**. Pauli hypothesized that a light neutral particle (a neutrino) is emitted from the nucleus simultaneously with the emission (or capture) of the electron. This neutrino carries away that portion of the disintegration energy not accounted for by the electron. In the case of K capture, it is clear that the neutrino must carry away substantially all the disintegration energy.

Present evidence suggests that the **neutrino mass** is measurably indistinguishable from zero. Thus, its speed, for any energy whatever, must always be that of light. Strong experimental evidence for the existence of the neutrino has recently been put forward by Reines and Cowan.*

* *Nature*, vol. 178, p. 449, 1956.

The neutrino concept formed the basis of the highly successful **Fermi theory of beta decay** proposed by Fermi in 1934. A principal prediction of the Fermi theory is a relationship giving the shape of the electron spectrum. It is, for so-called **allowed spectra**,

$$P(p) = \left(\frac{g^2}{2\pi^3 \hbar^7 c^3} \right) |M|^2 F(E, Z) p^2 (\hat{E} - E)^2 \quad (5-16)$$

In the above, g is a constant that specifies the strength of the interaction that produces the beta process. M is a **matrix element**; the transition probability for beta decay is proportional to the square of this quantity. $F(E, Z)$ is the so-called **Coulomb factor**. [See table of values by Rose, M. E., *et al.* in Seigbahn, K. (ed), "Beta and Gamma Ray Spectroscopy," Interscience, 1955.] It takes into account the fact that, after the electron has been emitted, it finds itself under the influence of the nuclear charge of the residual nucleus. Negative electrons will tend to be slowed down by this effect and positive ones speeded up. The Coulomb factor is usually significant only at low energies. $\hat{E}$ is the maximum energy of the distribution and p is the momentum of the emitted electron.

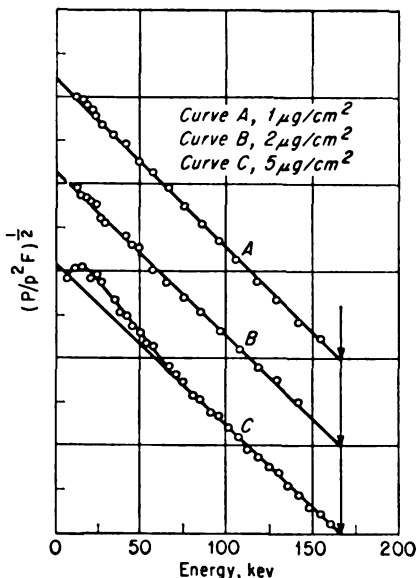


FIG. 5-8. Three Fermi plots for electrons from S-35. They are displaced vertically arbitrarily for convenience. The three curves were taken with different source thicknesses. Note that this is an important effect in determining whether the spectrum obeys Fermi's theory. (From Albert and Wu, *Phys. Rev.*, vol. 74, p. 847, 1948.)

Fermi plots are commonly made for the purpose of testing the nature of the distribution or finding the maximum energy. See Fig. 5-8.

The **disintegration constant** λ is related to $P(p)$ by

$$\lambda = \int_0^{\hat{p}} P(p) dt \quad (5-18)$$

Integrating yields

$$\lambda = |M|^2 \left(\frac{m^2 g^2 c^4}{2\pi^3 \hbar^7} \right) f(Z, \hat{E}) \quad (5-19)$$

where f is a function that can be calculated if the atomic number and the maximum energy of the beta process are known. The quantity f is called the **comparative**

If one plots

$$\left[\frac{P(p)}{p^2 F} \right]^{1/2} \quad (5-17)$$

as a function of energy E , a straight line should result if the electron spectrum obeys Eq. (5-16). Furthermore, the intercept of this straight line with the energy axis should be the maximum energy of the distribution. Such **Fermi**

lifetime of the beta emitter. The comparative lifetime is a device for eliminating the effects of maximum energy and nuclear charge in comparing the lifetimes of various beta emitters. The ft values (tabulation by Feingold, A. M., *Revs. Mod. Phys.*, vol. 23, p. 10, 1951) are direct measures of the strengths of the relative matrix elements causing the disintegration processes.

Figure 5-9 shows how the comparative lifetimes are distributed. It is clear that they cover a wide range of values and that the distribution is fairly continuous. Beta emitters with comparative lifetimes of the order of 10^4 sec are said to correspond to **allowed transitions**. Higher comparative lifetimes correspond to the so-called **first forbidden**, **second forbidden**, etc., transitions.

It is clearly not possible to classify a beta process uniquely entirely on the basis of its comparative lifetime; other experimental information is usually needed.

Gamma rays are electromagnetic radiations that travel through free space with the speed of visible light. They may be emitted directly from excited nuclei during nuclear reactions or accompanying beta or alpha decay. Figure 5-10, for example, shows some gamma rays accompanying the beta decay (by K capture) of Ir-192. (Baggerly, L. J., P. Marmier, F. Boehm, and J. W. M. DuMond, *Phys. Rev.*, vol. 100, p. 1364, 1955.) The gamma rays represent transitions between excited levels in the daughter nucleus Os-192. The energy measurements were made with the curved-crystal gamma-ray spectrometer developed by DuMond.

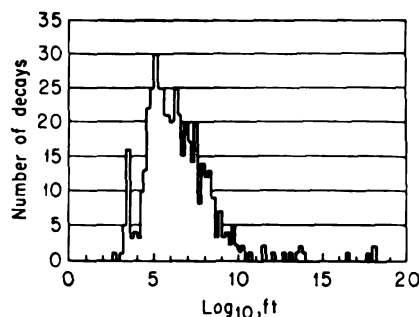


FIG. 5-9. The distribution of beta emitters as a function of comparative lifetime. (From Blatt and Weisskopf, "Theoretical Nuclear Physics," Wiley, 1952.)

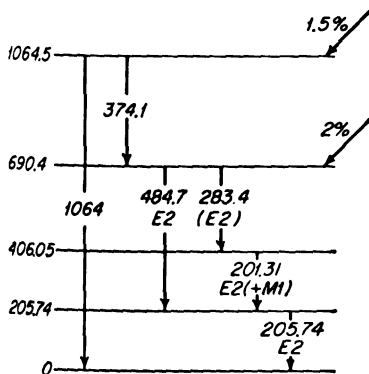


FIG. 5-10. Energy levels in Os-192 and some gamma transitions between them. The energies and spins of the levels are given as are the energies of the gamma rays (in kev). The intensities of the gamma rays are represented by the shading of the vertical lines. The upper two excited Os-192 levels are produced by the beta decay of Ir-192, 1.5 per cent of the decays producing the upper level and 2 per cent producing the level just below it.

a positive and negative electron disappear producing, in a framework in which the annihilating particles are at rest, two oppositely directed gamma rays, each of energy m_0c^2 (0.51 Mev).

Gamma rays arising from transitions between well-defined nuclear levels are classified as **dipole**, **quadrupole**, **octopole**, etc., transitions. In each of these, classes may be further characterized as either **electric** or **magnetic** radiation, for example, electric-dipole or magnetic-dipole radiation. This classification scheme is based on the nature of the matrix element whose square determines the disintegration probability for the process. For example, if the matrix element has the form

$$\int \psi_f^* z \psi_i dv$$

we are dealing with **electric-dipole** radiation. If it has the form

$$\int \psi_f^* (3z^2 - r^2) \psi_i dv$$

we are dealing with **electric-quadrupole** radiation.

The character of a given radiation can be determined experimentally by comparing the experimentally observed disintegration constant with those predicted by theory. For example, Blatt and Weisskopf have derived a theory which predicts the following disintegration constants: For electric radiation

$$\lambda_e \approx \lambda_0 \frac{4.4(l+1)}{l[(2l+1)!!]^2} \left(\frac{3}{l+3}\right) \left(\frac{E_\gamma}{E_0}\right)^{2l+1} \left(\frac{R}{R_0}\right)^{2l}$$

For magnetic radiation

$$\lambda_m \approx \lambda_0 \frac{1.9(l+1)}{l[(2l+1)!!]^2} \left(\frac{3}{l+3}\right) \left(\frac{E_\gamma}{E_0}\right)^{2l+1} \left(\frac{R}{R_0}\right)^{2l-2}$$

in which $\lambda_0 \approx 10^{21} \text{ sec}^{-1}$, $E_0 \approx 197 \text{ Mev}$, and $R_0 = 10^{-13} \text{ cm}$. The double factorial symbol means, for example,

$$7!! = 7.5.3.1$$

In these equations l , which takes on the values 1, 2, 3, etc., classifies the radiation as **dipole**, **quadrupole**, **octopole**, etc. The physical significance of l is that $l(h/2\pi)$ is the angular momentum carried away by the outgoing photon. Figure 5-11 shows a plot of $\log \tau(R/R_0)^6$ vs. energy for a number of gamma emitters. The fit to the theoretical curve predicted for electric-dipole radiation is quite good. For other classes of radiation the fit is not so good.

It is also possible to classify gamma rays by measuring other quantities such as, for example, **angular correlation** of successively emitted gamma rays.

Closely related to gamma emission is **internal conversion**. This is a process in which the excited nucleus delivers its excitation energy directly to an external electron which is ejected from its shell. These conversion electrons can be distinguished from true nuclear electrons by their discrete energies. The energy of a given conversion electron is given by

$$E_e = E_\gamma - E_x$$

where E_γ is the energy of the accompanying gamma ray and E_x is the binding energy of the shell from which the electron is ejected. Radioactive nuclei that decay by

internal conversion often provide a convenient source of monoenergetic electrons for calibrating nuclear instruments.

The **total conversion coefficient** is defined from

$$\alpha = \frac{R_e}{R_\gamma}$$

where R_e is the rate at which conversion electrons are produced and R_γ is the rate at which gamma rays are produced.

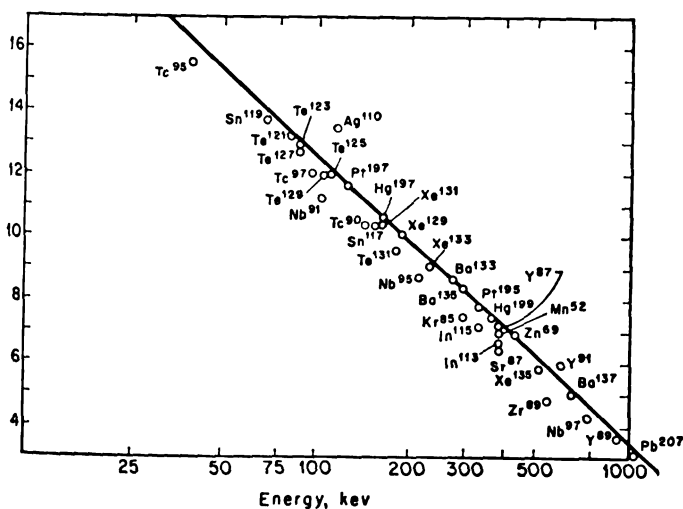


Fig. 5-11. A plot for some gamma emitters showing the fit to the theoretical curve expected for magnetic radiations with $l = 4$ (M4 transitions). τ is the mean life. (From Goldhaber and Sunyar, *Phys. Rev.*, vol. 83, p. 906, 1951.)

Partial conversion coefficients, referring to a particular electron shell, are defined from

$$\alpha = \frac{R_{ex}}{R_\gamma}$$

where R_{ex} is the rate at which electrons are ejected from the shell in question.

The partial conversion coefficients can be calculated from theory in terms of the atomic number, the energy, and the order of the transition. See the calculations of Rose and his collaborators [Siegbahn, K. (ed.), "Beta- and Gamma-ray Spectroscopy," p. 904, Interscience, 1955].

In a few cases **conversion electrons** are noted in which there are no accompanying gamma rays. For such cases the conversion coefficient is infinite. It always proves on examination that for these transitions both the initial and final nuclear states have zero angular momentum. Gamma emission between such states is strictly forbidden. This can be seen on the basis of a classical analogy to gamma emission. If the nucleus has zero angular momentum, it has complete spherical symmetry. The classical equivalent of the gamma emission in this case would be the radiation

emitted from a pulsating sphere. For points outside such a sphere, no electric or magnetic fields would be set up by the pulsating sphere. In other words, there would be no radiation. The external electrons on the other hand can, with a certain probability, be present inside this pulsating sphere. In other language, the electron and the nuclear wave functions overlap. Inside a pulsating sphere, time-varying electric and magnetic fields are set up, and the possibility for energy transfer does exist.

Section 6

SOURCES OF RADIATION

By

TRUMAN P. KOHMAN, Ph.D., *Professor of Chemistry, Carnegie Institute of Technology. (Natural Radioactivity)*

M. STANLEY LIVINGSTON, Ph.D., *Professor of Physics, Massachusetts Institute of Technology; Director, Cambridge Accelerator Project. (Particle Accelerators)*

THOMAS H. ROGERS, *Director of Engineering, Machlett Laboratories, Inc. (Electrical Sources of X-radiation)*

WILLIAM M. BREAZEALE, Ph.D., *Staff Consultant, Atomic Energy Division, the Babcock and Wilcox Company, Lynchburg, Virginia. (Nuclear Reactors)*

NAOMI A. HALLDEN, *Analytical Branch, Health and Safety Laboratory, U.S. Atomic Energy Commission, New York. (Radioactive Isotopes and Their Properties)*

Natural Radioactivity.....	6-2
Natural Radionuclides	6-2
Naturally Radioactive Materials.....	6-15
Particle Accelerators	6-33
Accelerator Types.....	6-33
Radiations from Accelerators.....	6-45
Shielding Arrangements.....	6-54
Ultra-high-energy Accelerators.....	6-59
Electrical Sources of X-radiation.....	6-61
X-ray Tubes	6-62
Tubes for Medical Therapy.....	6-64
Tubes for Industrial Applications.....	6-66
Electric Circuits for Energizing X-ray Tubes.....	6-68
Incidental X-ray Sources.....	6-71
X-ray Image Intensifiers.....	6-72
Nuclear Reactors.....	6-74
Reactor Physics.....	6-74
Reactor Classifications.....	6-78
Radioactive Isotopes and Their Properties.....	6-89

NATURAL RADIOACTIVITY

Truman P. Kohman

NATURAL RADIONUCLIDES

A classification of natural radionuclides can be made according to their mode of occurrence in nature, as follows: primary, secondary, induced, and extinct.

Primary natural radionuclides are unstable nuclides occurring naturally at the present time by virtue of very long lifetimes, so that detectable to substantial amounts have survived the interval since the elements were created. There is evidence that the elements of the solar system were formed $\sim 6 \times 10^9$ years ago. Consequently a nuclide must have a half-life of at least $\sim 3 \times 10^8$ years in order to have avoided complete decay. Nuclides with half-lives greater than $\sim 10^{10}$ years have undergone very little decay in the elapsed time. The most significant as radiation sources are those whose half-lives are similar to the elapsed time, especially

Table 6-1. Primary Natural Radionuclides

Nuclide	Natural isotopic abundance	Half-life, years	Mode of disintegration	Branching fraction	Specific activity in natural element, disintegration/sec/g
U ²³⁵	0.00720	7.0×10^8	α	1	570
K ⁴⁰	0.000118	1.27×10^9	$\left\{ \begin{array}{l} \beta^- \\ \epsilon \end{array} \right.$	$\left\{ \begin{array}{l} 0.892 \\ 0.108 \end{array} \right.$	$\left. \begin{array}{l} 28.0 \\ 3.40 \end{array} \right\} 31.4$
U ²³⁸	0.99274	4.51×10^9	α	1	12,240
Th ²³²	1	1.42×10^{10}	α	1	4,010
Lu ¹⁷⁶	0.0260	2.4×10^{10}	β^-	1	83
Rb ⁸⁷	0.2785	5.0×10^{10}	β^-	1	860
Re ¹⁸⁷	0.6293	6.2×10^{10}	β^-	1	720
La ¹³⁸	0.00089	1.2×10^{11}	$\left\{ \begin{array}{l} \beta^- \\ \epsilon \end{array} \right.$	$\left\{ \begin{array}{l} 0.37 \\ 0.63 \end{array} \right.$	$\left. \begin{array}{l} 0.27 \\ 0.46 \end{array} \right\} 0.73$
Sm ¹⁴⁷	0.1507	1.25×10^{11}	α	1	106
Pt ¹⁹⁰	0.00012	5.9×10^{11}	α	1	0.014
W ¹⁸⁰	0.00135	$\sim 3 \times 10^{14}$	α	1	~ 0.0003
V ⁵⁰	0.0025	$\sim 4 \times 10^{14}$	ϵ	1	~ 0.002
In ¹¹⁵	0.9577	$\sim 6 \times 10^{14}$	β^-	1	~ 0.18
Nd ¹⁴⁴	0.2387	$\sim 2 \times 10^{15}$	α	1	~ 0.011
Ce ¹⁴²	0.1107	$\sim 5 \times 10^{15}$	α	1	~ 0.002
Bi ²⁰⁹	1	$\sim 2 \times 10^{17}$	α	1	~ 0.0003

K-40 ($H = 1.27 \times 10^9$ years), U-238 ($H = 4.51 \times 10^9$ years), and Th-232 ($H = 14.2 \times 10^9$ years). Those of much shorter lifetime, such as U-235 ($H = 0.70 \times 10^9$ years), are weaker because of substantial decay; and those of much longer lifetime, such as Sm-147 ($H = 125 \times 10^9$ years), are weaker because of the smallness of their disintegration rates. All the known primary natural radionuclides are listed together with some of their properties in Table 6-1. U-235 and U-238 always occur together in natural uranium in the constant proportion given in Table 6-1. In natural uranium the ratio (U-235 activity)/(U-238 activity) = 0.047 ± 0.001 .

Secondary natural radionuclides are relatively short-lived nuclides that occur in nature as a result of continual formation by spontaneous transformation of primary natural radionuclides. The half-life might have any value up to $\sim 3 \times 10^8$ years, though the longest-lived known is U-234 ($H = 2.5 \times 10^5$ years). All that are known at present are members of the three disintegration series originating with U-238, U-235, and Th-232; they are listed, together with some of their properties, in Tables 6-2, 6-3, and 6-4, respectively. In undisturbed intact minerals of moderate geological age secular equilibrium prevails, and all members of a given series occur together at equal activity levels except where their concentrations are reduced by branching disintegration of an antecedent. In uranium-containing minerals the

Table 6-2. Secondary Natural Radionuclides Descended from U-238 (Uranium or Radium Series)

Nuclide	Half-life	Relative equilibrium activity	Mode of disintegration	Branching fraction
Th ²³⁴ (UX ₁)	24.1 days	1	β^-	1
Pa ^{234m} (UX ₂)	1.18 min	1	β^-	0.9937
Pa ^{234g} (UZ)	6.66 hr	0.0063	I.T.	0.0063
U ²³⁴ (UII)	248,000 years	1	β^-	1
Th ²³⁰ (Io)	80,000 years	1	α	1
Ra ²²⁶ (Ra)	1,620 years	1	α	1
Em ²²² (Rn)	3.825 days	1	α	1
Po ²¹⁸ (RaA)	3.05 min	1	β^-	0.00022
			α	0.99978
At ²¹⁸ (RaA')	~ 2 sec	0.00022	β^-	~ 0.001
			α	0.999
Em ²¹⁸ (RaA'')	1.3 sec	~ 0.0000002	α	1
Pb ²¹⁴ (RaB)	26.8 min	0.99978	β^-	1
Bi ²¹⁴ (RaC)	19.7 min	1	β^-	0.9996
			α	0.0004
Po ²¹⁴ (RaC')	0.00016 sec	0.9996	α	1
Tl ²¹⁰ (RaC'')	1.32 min	0.0004	β^-	1
Pb ²¹⁰ (RaD)	19.4 years	1	β^-	1
Bi ²¹⁰ (RaE)	5.01 days	1	β^-	0.9999983
			α	0.0000017
Po ²¹⁰ (RaF)	138.40 days	0.9999983	α	1
Tl ²⁰⁶ (RaE'')	4.19 min	0.0000017	β^-	1

Table 6-3. Secondary Natural Radionuclides Descended from U-235 (Actinium Series)

Nuclide	Half-life	Relative equilibrium activity	Mode of disintegration	Branching fraction
Th ²³¹ (UY)	25.6 hr	1	β^-	1
Pa ²³¹ (Pa)	34,300 years	1	α	1
Ac ²²⁷ (Ac)	21.6 years	1	β^-	0.988
			α	0.012
Th ²²⁷ (RdAc)	18.4 days	0.988	α	1
Fr ²²³ (AcK)	21 min	0.012	β^-	0.99994
			α	0.00006
Ra ²²³ (AcX)	11.7 days	0.9999993	α	1
At ²¹⁹	0.9 min	0.0000007	β^-	0.03
			α	0.97
Em ²¹⁹ (An)	3.92 sec	0.9999993	α	1
Bi ²¹⁵	8 min	0.0000007	β^-	1
Po ²¹⁵ (AcA)	0.00183 sec	1	β^-	0.000002
			α	0.999998
At ²¹⁵	~ 0.0001 sec	0.000002	α	1
Pb ²¹¹ (AcB)	36.1 min	0.999998	β^-	1
Bi ²¹¹ (AcC)	2.16 min	1	β^-	0.0032
			α	0.9968
Po ²¹¹ (AcC')	0.52 sec	0.0032	α	1
Tl ²⁰⁷ (AcC'')	4.79 min	0.9968	β^-	1

Table 6-4. Secondary Natural Radionuclides Descended from Th-232 (Thorium Series)

Nuclide	Half-life	Relative equilibrium activity	Mode of disintegration	Branching fraction
Ra ²²⁸ (MsTh ₁)	6.7 years	1	β^-	1
Ac ²²⁸ (MsTh ₂)	6.13 hr	1	β^-	1
Th ²²⁸ (RdTh)	1.910 years	1	α	1
Ra ²²⁴ (ThX)	3.64 days	1	α	1
Em ²²⁰ (Tn)	52 sec	1	α	1
Po ²¹⁶ (ThA)	0.158 sec	1	α	1
Pb ²¹² (ThB)	10.64 hr	1	β^-	1
Bi ²¹² (ThC)	60.5 min	1	β^-	0.646
			α	0.354
Po ²¹² (ThC')	0.30 μ sec	0.646	α	1
Tl ²⁰⁸ (ThC'')	3.10 min	0.354	β^-	1

U-235 and U-238 series occur together, the latter at considerably lower activity levels. In such materials (including uranium minerals) in which secular equilibrium prevails the U-235/U-238 activity ratio determines the relative activities of the members of each series; the relative equilibrium activities given in Table 6-3 should be multiplied by 0.047 for comparison with the corresponding figures in Table 6-2. Among the important secondary natural radionuclides are Th-230 (ionium), Ra-226 ("radium"), Em-222 (radon), and Po-210 ("polonium") in the U-238 series and Ra-228 (mesothorium I) in the Th-232 series.

Induced natural radionuclides are relatively short-lived nuclides that are found in nature as a result of continual production by nuclear reactions. The responsible nuclear reactions may be initiated by the primary radiations or secondary neutrons in radioactive minerals (producing Pu-239, Np-237, Cl-36, Sr-90, and other fission products in uranium minerals; U-233 in thorium minerals); or by primary and secondary cosmic-ray particles in the earth's atmosphere (H-3, Be-7, Be-10, C-14, P-32, P-33, Na-22, S-35, Cl-39), at the earth's surface (Cl-36, Co-60), or in extra-terrestrial objects such as meteorites (H-3, Be-10, Al-26). The best known are natural radiocarbon (C-14) and natural tritium (H-3). In all cases, the activities are extremely low and can be observed only by sensitive instruments.

Extinct natural radionuclides are hypothetical unstable nuclides whose lifetimes are too short for detectable amounts to have persisted from the time of element

Table 6-5. Gamma Radiations of the Primary Natural Radionuclides

Nuclide	Gamma energy, Mev	No. of photons per disintegration	No. of photons/sec/g natural element
U ²³⁸	0.048	Transition largely internally converted; photon emission insignificant	
U ²³⁵	~0.18 ~0.37 Others(?)	~0.7 ~0.05 (Small)	~400 ~30 (Small)
Th ²³²	0.059	Transition largely internally converted; photon emission insignificant	
K ⁴⁰	1.46	0.108	3.40
V ⁵⁰	1.5 ₈	1	~0.001 ₆
La ¹³⁸	0.81 1.43	0.37 0.63	0.27 0.46
Lu ¹⁷⁶	0.089 0.203 0.306	0.16 0.85 0.94	13 70 78

Table 6-6. Gamma Radiations of the Secondary Natural Radionuclides Descended from U-238

Nuclide	Gamma energy, Mev	No. of photons per disintegration	Relative No. of photons in equilibrium mixture
Th^{234} (UX ₁)	0.029	~0.004	~0.004
	0.063	~0.03	~0.03
	0.092	~0.04	~0.04
$\text{Pa}^{234\text{m}}$ (UX ₂)	0.255	<0.0010	<0.0010
	0.770	≤0.0087	≤0.0087
	1.01	0.015	0.015
	1.24	(Small)	(Small)
	1.44	0.0009	0.0009
	1.69	0.0007	0.0007
	1.83	0.0011	0.0011
$\text{Pa}^{234\text{g}}$ (UZ)	0.125	<0.15	<0.0009
	0.225	0.07 ₅	0.0004 ₇
	0.293	(Small)	(Very small)
	0.368	0.02 ₇	0.0001 ₇
	0.566	0.06 ₂	0.0003 ₉
	0.732	0.09	0.0006
	0.877	0.14	0.0009
	0.924	0.25	0.0016
	1.24	(Small)	(Very small)
	1.43	0.035	0.00022
	1.68	0.025	0.00016
U^{234} (U _{II})	0.052	Transition largely internally converted; photon emission insignificant (Very small) (Very small)	
	0.117(?)		
Th^{230} (Io)	0.068	0.0059	0.0059
	0.142	0.00070	0.00070
	0.184	0.00014	0.00014
	0.253	0.00017	0.00017
Ra^{226} (Ra)	0.187	0.030	0.030
	0.260	0.00007 ₅	0.00007 ₅
Em^{222} (Rn)	0.510	0.0007	0.0007
Pb^{214} (RaB)	0.053	0.016	0.016
	0.206	~0.002	~0.002
	0.242	0.071	0.071
	0.259	~0.002	~0.002
	0.275	~0.002	~0.002
	0.295	0.20	0.20
	0.352	0.36	0.36
	0.480	~0.003	~0.003
	0.534	~0.002	~0.002
	0.780	~0.005	~0.005
Bi^{214} (RaC) (α branch)	0.062	≤0.46	≤0.0002
	0.191	≤0.16	≤0.0001

Table 8-6. Gamma Radiations of the Secondary Natural Radionuclides
Descended from U-238 (Continued)

Nuclide	Gamma energy, Mev	No. of photons per disintegration	Relative No. of photons in equilibrium mixture
Bi ²¹⁴ (RaC) (β branch)	0.426	0.036	0.036
	0.450	0.016	0.016
	0.496	0.056	0.056
	0.609	0.295	0.295
	0.652 } 0.661 }	0.024	0.024
	0.703	0.012	0.012
	0.769	0.044	0.044
	0.787	0.011	0.011
	0.806	0.011	0.011
	0.821	0.008	0.008
	0.935	0.033	0.033
	1.120	0.131	0.131
	1.155	0.025	0.025
	1.207	0.012	0.012
	1.238	0.056	0.056
	1.281	0.016	0.016
	1.378	0.046	0.046
	1.385 } 1.391 } 1.396 }	0.046	0.046
	1.402 } 1.408 } 1.438 }		
	1.509	0.027	0.027
	1.583	0.013	0.013
	1.605	0.004 ₄	0.004 ₄
	1.728	0.018	0.018
	1.764	0.160	0.160
	1.784	0.028	0.028
	1.791	0.009	0.009
	1.848	0.018	0.018
	1.862	0.014	0.014
	1.900	0.012	0.012
	2.017	0.003 ₈	0.003 ₈
	2.085	0.005 ₇	0.005 ₇
	2.117	0.015	0.015
	2.204	0.063 ₆	0.063 ₆
	2.29	0.004 ₄	0.004 ₄
	2.34	0.002 ₆	0.002 ₆
	2.432	0.026 ₀	0.026 ₀
	2.72	0.0017	0.0017
	2.89	0.0003	0.0003
	3.03	0.0008	0.0008
Po ²¹⁴ (RaC')	0.458(?)	Unconverted gamma has not been observed	
Tl ²¹⁰ (RaC'')	0.297	~1	~0.0004
	0.783	~1	~0.0004
	1.1(?)	(Small)	(Very small)
	1.3(?)	(Small)	(Very small)
	2.36	~1	~0.0004
Pb ²¹⁰ (RaD)	0.0467	0.045	0.045
Po ²¹⁰ (RaF)	0.804	0.000012	0.000012

**Table 6-7. Gamma Radiations of the Secondary Natural Radionuclides
Descended from U-235**

Nuclide	Gamma energy, Mev	No. of photons per disintegration	Relative No. of photons in equilibrium mixture
Th ²³¹ (UY)	0.025	~0.7	~0.7
	0.059	~0.16	~0.16
	0.084	~0.4	~0.4
	~0.107	~0.03	~0.03
	~0.122	~0.008	~0.008
	0.165	~0.007	~0.007
	~0.210	~0.001	~0.001
	~0.230	~0.0004	~0.0004
Pa ²³¹ (Pa)	0.027	~0.05	~0.05
	~0.05	~0.002	~0.002
	~0.10	≤0.01	≤0.01
	~0.30	~0.05	~0.05
Th ²²⁷ (RdAc)	0.050	0.15	0.15
	0.057	(Small)	(Small)
	0.080	~0.1	~0.1
	0.100	(Small)	(Small)
	0.113	(Small)	(Small)
	0.21	~0.2	~0.2
	0.24		
	0.26		
Others	Individually small, but together may be appreciable		
Fr ²²³ (AcK)	0.050	0.40	0.0048
	0.080	0.24	0.0029
	0.215	0.03 ₆	0.0004
	0.310	~0.008	~0.00010
Ra ²²³ (AcX)	0.122	0.02	0.02
	0.144	0.041	0.041
	0.154	0.055	0.055
	0.180	0.005	0.005
	0.270	0.10	0.10
	0.324	0.023	0.023
	0.338	0.028	0.028
	0.371	0.0045	0.0045
Em ²¹⁹ (An)	0.272	0.086	0.086
	0.401	0.048	0.048
Pb ²¹¹ (AcB)	0.065	~0.1	~0.1
	0.083		
	0.404		
	0.425		
	0.487		
	0.764		
	0.829	0.054	0.054
Bi ²¹¹ (AcC)	0.351	0.14	0.14
Po ²¹¹ (AcC')	0.56	~0.005	~0.00001 ₆
	0.87	~0.005	~0.00001 ₆
Tl ²⁰⁷ (AcC'')	0.87	~0.05	~0.05

**Table 6-8. Gamma Radiations of the Secondary Natural Radionuclides
Descended from Th-232**

Nuclide	Gamma energy, Mev	No. of photons per disintegration	Relative No. of photons in equilibrium mixture
Ac ²²⁸ (MsTh ₂)	0.057	~0.003	~0.003
	0.098	(Moderate)	(Moderate)
	0.128	0.04	0.04
	0.155	(Very small)	(Very small)
	0.184	(Very small)	(Very small)
	0.209	0.045	0.045
	0.270	0.03	0.03
	0.328	0.04	0.04
	0.336	0.11	0.11
	0.410	~0.04	~0.04
	0.458	~0.03	~0.03
	0.779	~0.008	~0.008
	0.790	~0.06	~0.06
	0.831	~0.016	~0.016
	0.908	0.25	0.25
	0.960 } 0.966 }	0.20	0.20
	1.587	(Moderate)	(Moderate)
	Others	Individually small, but together may be appreciable	
Th ²²⁸ (RdTh)	0.084	0.016	0.016
	0.133	0.0026	0.0026
	0.169	0.0009	0.0009
	0.205	0.0003	0.0003
	0.217	0.0027	0.0027
Ra ²²⁴ (ThX)	0.241	0.038	0.038
	0.29	~0.00008	~0.00008
	0.41	~0.00004	~0.00004
	0.65	~0.00009	~0.00009
Em ²²⁰ (Tn)	0.542	~0.0002 ₆	~0.0002 ₆
Pb ²¹² (ThB)	0.115	0.004	0.004
	0.239	0.40	0.40
	0.300	0.028	0.028
	0.415	0.02	0.02
Bi ²¹² (ThC) (α branch)	0.040	0.034	0.012 ₀
	0.287 } 0.327 }	0.012	0.004 ₂
	0.431 } 0.471 }	0.0013	0.0004 ₆
	0.451	0.010	0.03 ₄

Table 6-8. Gamma Radiations of the Secondary Natural Radionuclides Descended from Th-232 (Continued)

Nuclide	Gamma energy, Mev	No. of photons per disintegration	Relative No. of photons in equilibrium mixture
Bi ²¹² (ThC) (β branch)	0.729	0.09 ₃	0.06 ₀
	0.83	0.09 ₆	0.06 ₂
	1.03	0.02 ₆	0.01 ₇
	1.35	0.02 ₀	0.01 ₃
	1.50	0.02 ₀	0.01 ₃
	1.60	0.05 ₅	0.03 ₅
	1.67(?)	(Small)	(Small)
	1.80	0.03 ₄	0.02 ₂
	2.20	0.02 ₈	0.01 ₈
Tl ²⁰⁸ (ThC'')	0.233(?)	(Small)	(Small)
	0.252	(Small)	(Small)
	0.277	~ 0.06	~ 0.02
	0.511	0.3 ₀	0.11
	0.583	0.8 ₀	0.28
	0.763	(Small)	(Small)
	0.859	0.1 ₀	0.03 ₅
	1.094	~ 0.006	~ 0.002
	2.614	0.99 ₇	0.353

formation, yet long enough for their decay to have produced effects in nature which can be identified at the present time. The half-life would have to be in the range $\sim 3 \times 10^7$ years to $\sim 3 \times 10^8$ years, and certain other conditions would have to be met. No example is known as yet, and obviously any surviving effects would not include radiation.

From the viewpoint of radiation hazards, shielding calculations, counter backgrounds, etc., the **gamma radiations of the natural radionuclides** are often of interest. Since data on the intensities of these radiations, and to some extent also on the energies, are generally unavailable, a compilation is presented in Tables 6-5, 6-6, 6-7, and 6-8 for the primary natural radionuclides and the U-238, U-235, and Th-232 families, respectively. These tables are incomplete and the data to some extent uncertain because of the complexity of the radiations and the difficulty and present inadequacy of their experimental investigation.

In the preparation of Tables 6-1 to 6-8, a number of **sources of nuclear data** were used. Most of the data are to be found in various nuclear-data summaries, including the Table of Isotopes by Hollander, Perlman, and Seaborg (*Revs. Modern Phys.*, vol. 25, pp. 469-651, 1953), which is essentially complete through 1952; a revision of the same by Strominger, Hollander, and Seaborg (*Revs. Modern Phys.*, vol. 30, pp. 585-904, 1958), which is essentially complete through 1957; the "New Nuclear

Data" extracts completed by the Nuclear Data Group of the U.S. National Research Council and published as follows:

- 1952 Cumulation—*Nuclear Science Abstracts*, vol. 6, no. 24B (Dec. 31, 1952)
- 1953 Cumulation—*Nuclear Science Abstracts*, vol. 7, no. 24B (Dec. 31, 1953)
- 1954 Cumulation—*Nuclear Science Abstracts*, vol. 8, no. 24B (Dec. 31, 1954)
- 1955 Cumulation—*Nuclear Science Abstracts*, vol. 9, no. 24B (Dec. 31, 1955)
- 1956 Cumulation—*Nuclear Science Abstracts*, vol. 10, no. 24B (Dec. 31, 1956)

and the "New Nuclear Data" cards prepared and mailed monthly by the same group. In practically all cases, however, the original publications have been consulted, particularly in cases of conflicting reports. (See also p. 6-89.)

The **gamma radiations of Ra-226** with its disintegration products ("radium") make a special case, important because of the extensive use of radium as a radiation source and standard. It is a difficult problem because of the complexity of the radiations, the incompleteness of the experimental information, and the many inconsistencies among the various published experimental results. A new analysis of the data was made to obtain the intensities of the gamma radiations of Ra-226, Rn (Em-222), RaB (Pb-214), RaC (Bi-214), and RaC' (Tl-210), as follows.

For **Ra-226**, the alpha branching fraction to the 0.187-Mev state is 0.057 ± 0.003 (Asaro and Perlman, *Phys. Rev.*, vol. 88, p. 129, 1952). The fraction of all disintegrations leading to conversion electrons from that state is taken to be 0.027 ± 0.003 , the average of 0.030 ± 0.003 (Victor, Teillac, Falk-Vairant, and Bouissières, *J. phys. radium*, ser. 8, vol. 13, p. 565, 1952), 0.0269 (Roy and Goes, *Compt. rend.*, vol. 238, p. 469, 1954), 0.021 (Jurić and Stanojević, *Bull. Inst. Nuclear Sci. "Boris Kidrich" Belgrade*, vol. 5, p. 15, 1955), and 0.028 ± 0.005 (Albouy, *Ann. phys.*, ser. 13, vol. 1, p. 99, 1956). By difference, the intensity of unconverted 0.187-Mev gammas is 0.030 ± 0.004 per disintegration. According to Harbottle, McKeown, and Scharff-Goldhaber (*Phys. Rev.*, vol. 103, p. 1776, 1956), the ratio of intensities of 0.260- to 0.187-Mev gammas is 0.0025; accordingly, the intensity of the 0.260-Mev gamma is 0.00007₆ photons per disintegration. (These same authors found no evidence for the gammas of other energies reported by others.)

For **Rn (Em-222)**, the single datum tabulated is from Madansky and Rasetti (*Phys. Rev.*, vol. 102, p. 464, 1956).

For **RaB (Pb-214)**, the intensity of the 0.053-Mev gamma is from Tsien (*Compt. rend.*, vol. 217, p. 599, 1943). For the 0.242-Mev transition, Nielsen, Nielsen, and Waggoner (*Nuclear Phys.*, vol. 2, p. 476, 1956) give as the intensity of K-conversion electrons 0.045 ± 0.004 and as the most probable value of the K-conversion coefficient 0.63, whence the gamma intensity is 0.071. The relative intensities of the 0.242-, 0.295-, and 0.352-Mev gammas are given by Muller, Hoyt, Klein, and DuMond (*Phys. Rev.*, vol. 88, p. 775, 1952) as 0.2, 0.55, and 1.00, respectively; this gives 0.20 and 0.36 for the absolute intensities of the 0.295- and 0.352-Mev gammas. (The sum of these three gamma intensities, 0.63, is between the limits 0.61 and 0.73 deduced by Nielson *et al.*) Nielson *et al.* also give the relative intensities of the 0.242-, 0.480-, and 0.780-Mev gammas as 1, ~ 0.04 , and ~ 0.07 ; accordingly, the absolute intensities are ~ 0.003 for the 0.480-Mev gamma and ~ 0.005 for the 0.780-Mev gamma. The latter two transitions and four others of 0.206, 0.259,

0.275, and 0.534 Mev have a combined intensity of ~ 0.02 according to Nielson *et al.*; allowing something for conversion, and lacking more information, we arbitrarily assign ~ 0.002 to the unconverted gammas of each of the other four transitions.

For the alpha disintegration of **RaC (Bi-214)**, whose branching fraction is 0.0004, conversion electrons attributed to transitions of 0.062 and 0.191 Mev have been observed (Cork, Branyan, Stoddard, Keller, LeBlanc, and Childs, *Phys. Rev.*, vol. 83, p. 681, 1951). From the alpha-particle spectrum (Chang, *Phys. Rev.*, vol. 74, p. 1195, 1948) it can be deduced that the 0.062- and 0.191-Mev levels are populated in, respectively, ~ 0.46 and ~ 0.16 of the alpha disintegrations. These are upper limits to the frequencies of unconverted gammas. (In the following analysis these gammas are neglected.)

For **RaC' (Tl-210)**, which is involved in 0.0004 of the RaC disintegrations, the finding of Mayer-Kuckuk (*Z. Naturforsch.*, vol. 11a, p. 627, 1956) that practically all the disintegrations involve a cascade of 2.37-, 0.29-, and 0.783-Mev gammas has been strengthened by Daniel (*Z. Naturforsch.*, vol. 12a, p. 194, 1957).

For the beta disintegration of **RaC (Bi-214)**, whose branching fraction is 0.9996, the most complete and evidently most accurate relative intensities of the gammas from 0.609 to 2.432 Mev are those of Dzelpov and Sestapalova (*Nuovo cimento, Suppl.* 1, 1° Sem., ser. 10, vol. 3, p. 54, 1956); the intensities are all given relative to that of the 2.204-Mev gamma. Mann and Ozeroff (*Can. J. Research*, vol. 27A, p. 164, 1949) give the intensities of 0.426-, 0.450-, and 0.496-Mev gammas relative to the 0.609-Mev one. Daniel (*Z. Naturforsch.*, vol. 12a, p. 194, 1957) has reported additional gammas of 2.72, 2.89, and 3.03 Mev, and determined their intensity relative to that of 2.204 Mev. A single absolute intensity measurement exists: Backenstoss and Wohlleben (*Z. Naturforsch.*, vol. 10a, p. 384, 1955) give the intensity of the 2.43-Mev photon as 0.0248 ± 0.0012 per disintegration.

As a first step in the calculation of the **intensities of the RaC gamma radiations**, the above-mentioned relative values of all these gammas were converted to a tentative absolute basis by adopting the above-mentioned absolute value for the 2.43-Mev gamma.

Then the combined "gamma-ray output" of Ra + Rn + RaB + RaC + RaC' through 0.500 mm Pt was calculated as follows. Transmission factors in this thickness of platinum (1.0725 g/cm^2) were computed for each energy using mass-absorption coefficients obtained by interpolation among the theoretical values of Davisson and Evans (*Revs. Modern Phys.*, vol. 24, p. 79, 1952) and Gladys White Grodstein [*Natl. Bur. Standards (U.S.) Circ.* 583, 1957, revision of Gladys R. White, *Natl. Bur. Standards (U.S.) Rept.* 1003, 1952]. Air-dose "build-up factors" for each energy in the same thickness of platinum were obtained by a triple-interpolation procedure using the values calculated for several other elements by Goldstein and Wilkins (*AEC Rept.* NYO-3075, 1954). Linear energy-absorption coefficients ("true" absorption coefficients) for air in the region up to 1 Mev were obtained from White [*Natl. Bur. Standards (U.S.) Rept.* 1003, Table II, 1952]; for energies above 1.022 Mev these coefficients were recalculated so as to allow for the escape of annihilation radiation in pair-production interactions (so as to be consistent with the way the build-up factors are calculated), using the "true absorption" cross sections for the Compton process given by White [*Natl. Bur. Standards (U.S.) Rept.* 1003,

Table 4, 1952] and pair-production cross sections given by Grodstein [*Natl. Bur. Standards (U.S.) Circ.* 583, Tables 15, 16, 23, 1957]. For each photon, its contribution to the energy transferred per Ra-226 disintegration to a 1-cm-thick shell of dry air at standard temperature and pressure (STP) was obtained as the product of: the photon energy; its intensity in photons per disintegration; its transmission factor; its build-up factor; the energy-absorption coefficient of air for that energy. The contributions of all gammas were added, the result being $(0.600 + q \times 4.747) \times 10^{-5}$ Mev/cm·disintegration. Here, q is an adjustment factor for RaC, the value of which would be unity if the absolute intensity of the 2.43-Mev gamma were 0.0248 photon/disintegration. The gamma-ray output of radium in 0.500 mm Pt, in units of roentgens per hour·gram at 1 m, is then given by

$$\begin{aligned} & (3.61 \times 10^{10} \text{ dis/sec} \cdot \text{g})[(0.600 + q \times 4.747) \times 10^{-5} \text{ Mev/cm} \cdot \text{dis}] \\ & (3,600 \text{ sec/hr})(4\pi)^{-1}(100 \text{ cm})^{-2}(7.08 \times 10^4 \text{ Mev/cm}^3 \cdot \text{r})^{-1} \\ & = (0.088 + q \times 0.693) \text{ r/hr} \cdot \text{g at 1 m} \end{aligned}$$

Here 3.61×10^{10} disintegrations/sec·g is the specific activity of radium (Kohman, Ames, and Sedlet, *Natl. Nuclear Energy Ser.*, vol. IV-14B, p. 1675, 1949; Sebaoun, *Ann. phys.*, ser. 13, vol. 1, p. 680, 1956); 7.08×10^4 Mev/cm³·r is the energy transfer per unit volume of STP air per roentgen, corresponding to an ionization energy of 34.0 ev per ion pair (the best present value according to Weiss and Bernstein, *Radiation Research*, vol. 6, p. 603, 1957).

The value of the adjustment factor q could then be obtained by comparison of the result of this calculation with the measured value of the **gamma-ray output of radium through 0.5 mm Pt**. Evidently the best value is that of Taylor and Singer (*Am. J. Roentgenol. Radium Therapy*, vol. 44, p. 428, 1940), who obtained 0.816 ± 0.004 r/hr·g at 1 m through 0.500 mm Pt, using a free-air ionization chamber. Measurements with cavity chambers give confirmation; the average of several published results, when properly corrected for wall effects, is 0.82 r/hr·g at 1 m (Whyte, *Nucleonics*, vol. 12, no. 2, p. 18, 1954). Thus,

$$q = \frac{0.816 \text{ r/hr} \cdot \text{g at 1 m} - 0.088 \text{ r/hr} \cdot \text{g at 1 m}}{0.693 \text{ r/hr} \cdot \text{g at 1 m}} = 1.050$$

The tentative RaC gamma intensities referred to above were each multiplied by this value of q to give the final intensities given in Table 6-6. (The resulting value of 0.0260 photon per disintegration for the 2.43-Mev gamma differs by only one standard deviation from the directly determined value 0.0248 ± 0.0012 of Backenstoss and Wohlleben. This is quite satisfactory agreement, although it suggests that the relative intensities of Dzelpov and Sestapalova for the low-energy gammas might be systematically low by several per cent. On the other hand, our value for the intensity ratio of the RaC 0.609-Mev and RaB 0.352-Mev gammas, 0.83, disagrees markedly with the value deduced from a direct comparison by Muller *et al.*, 1.6.)

A parallel calculation to that described above, but with omission of absorption and build-up factors, was made in order to evaluate the effect of 0.500 mm Pt and the **gamma-ray output of unshielded radium**. The result for the total energy transfer to air was 6.014×10^{-5} Mev/cm·disintegration, and for the gamma-ray output

of unshielded radium 0.878 r/hr·g at 1 m. The resulting transmission factor (air-dose basis) of 0.500 mm Pt is 0.929. [This is in excellent agreement with an experimental value of $(1.072)^{-1} = 0.934$ obtained by Taylor and Singer (*Am. J. Roentgenol. Radium Therapy*, vol. 44, p. 428, 1940) from the slope of a Pt absorption curve between 0.5 and 1.4 mm.]

In these tables and calculations, the **X-radiation of the heavy natural radionuclides** has been neglected. Although prominent in many of these nuclides as a result of internal conversion of gamma radiation, it is negligible as a hazard in comparison with the associated hard gamma radiation. It is also quantitatively insignificant for the case of radium in 0.500 mm of Pt; the *K* X-rays of the elements involved have energies only moderately above the *K*-absorption edge of Pt and are >99.4 per cent absorbed by this thickness; the *L* X-rays are even more completely absorbed. According to Cederlund (*Arkiv Fysik*, vol. 11, p. 431, 1957), such sources emit 0.030 photon of ~65-keV energy (mostly Pt *K* X-rays) per radium disintegration. Their contribution of 0.18 per cent to the total output is automatically included in the above calculated value because of the use of build-up factors. However, the X-rays of heavy elements would make an appreciable additional contribution to the output of electromagnetic radiation from unshielded radium, and the actual transmission factor for 0.500 mm Pt would be significantly smaller than the above values for the gamma radiation alone.

There is considerable variance in the importance of various **natural radionuclides as radiation sources**. In their natural state, gamma radiation never constitutes a health hazard, except possibly in rich uranium mineral deposits. In both the uranium and thorium series, the bulk of the gamma energy is emitted by the secondary members of the chains, and in general the hazards increase with the degree of enrichment of these disintegration products. Thus, radium-enriched tailings from uranium-ore processing may create a hazardous gamma-ray field. (Some such residual slimes are sufficiently active to be useful as marking compounds in oil-well casings.) Highly enriched radium (Ra-226) and mesothorium (Ra-228), which always have associated with them disintegration products emitting strong gamma radiation, constitute hazardous gamma sources when present in millicurie or greater quantities without adequate shielding. Practically all the gamma radiation associated with radium comes from short-lived "active-deposit" products following radon (Em-222). Consequently, when the radon is removed from a source of radium it acquires nearly all the gamma activity as its short-lived products grow in toward equilibrium. If the radon is allowed to escape, the gamma radiation remains behind with the "active deposit"; the activity decays away, however, in a matter of hours. The gamma activity associated with the long-lived decay products of radium and radon, RaD (Pb-210), RaE (Bi-210), and RaF (Po-210), is negligible from the viewpoint of radiation hazard. The gamma activity of freshly prepared mesothorium (Ra-228) is rather low but increases slowly to a maximum in several years and eventually decays with the Ra-228 half-life. The gamma radiations from natural radionuclides even in their normal concentrations in the ground or in building materials can make significant contributions to the backgrounds of nuclear-radiation detectors, particularly low-level counters. The natural radioactivity of potassium, due to K-40, never constitutes a health hazard, but its gamma activity is important in connection with the backgrounds of low-level radiation detectors and of counter

tubes made of glass. All the other primary natural radionuclides, as well as all the induced natural radionuclides, are insignificant even in respect to radiation backgrounds.

NATURALLY RADIOACTIVE MATERIALS

The **gamma radiation of natural materials** is due mainly to K-40, U-238, U-235, Th-232, and the decay products of the latter three. Because of the considerably greater specific activity and gamma-energy release per disintegration of natural uranium and thorium as compared with potassium, uranium and thorium minerals are much stronger gamma-radiation sources than potassium minerals. Potassium minerals are, in turn, much stronger gamma sources than the minerals of any of the remaining elements.

Uranium and thorium are commonly associated in minerals; accordingly **uranium and thorium minerals** are listed together in Tables 6-9 and 6-10. The first lists minerals containing one or both of these elements as major constituents; the second lists minerals with minor amounts of one or both. The information in these two tables has been taken from Frondel, J. W., and M. Fleischer, *Glossary of Uranium- and Thorium-bearing Minerals*, *U.S. Geol. Survey Bull.* 1009-F, 3d ed., 1955. This report gives additional information about the occurrence and characteristics of most of the minerals and references to more detailed sources of information.

Table 6-11 lists the **potassium minerals** of highest potassium contents and those of most common occurrence, together with information about their content of potassium. The designation "nom." means that the potassium content has simply been calculated from the nominal chemical formula given. In such cases in particular, but also in other cases, the actual potassium content is frequently somewhat lower because of admixture of impurities and, especially in the case of silicates, because of isomorphous replacement of potassium by other elements. In general, the potassium content of minerals of a given kind may vary considerably. The authority for most of the material in this table is "An Index of Mineral Species and Varieties Arranged Chemically."

The **radioactivity of rocks** is due almost entirely to their contents of potassium, uranium, and thorium. The average abundances of these elements at the earth's surface happen to be roughly in inverse proportion to their specific gamma activities (when U and Th are in equilibrium with their active disintegration products); so that the three elements are of approximately equal importance in their contribution to radiation backgrounds in most surroundings. The ratio of rubidium to potassium is fairly constant, and about equal to one-ninetieth on a weight basis, in most terrestrial rocks; although its specific activity is about twenty-seven times that of potassium, it is a pure beta emitter and does not contribute to general radiation backgrounds.

All these quantitatively most important radioelements have been concentrated in the upper layers of the earth's crust, as is shown by comparison of their average abundances in crustal rocks (K, 2.6 per cent; U, ~ 3 ppm; Th, ~ 12 ppm) with those in **chondrite meteorites** [K, 0.09 per cent; U, 0.01 ppm; Th, ~ 0.04 ppm (?)]. However, the variations in the concentrations of individual elements in particular rock types are even greater than is the case in minerals of a given type. In the case

Table 6-9. Uranium and Thorium Minerals

Name	Nominal composition; comments	Nominal, typical, or average content or range	
		U, %	Th, %
Abernathyite	$K_2(UO_2)_2(AsO_4)_2 \cdot 8H_2O$	52.8	
Ampangabeite	$(Y, Er, U, Ca, Th)_2(Nb, Ta, Fe, Ti)_7O_{18}?$	17.1	1.8
Hydroeuxenite.....	Synonym validity questionable		
Andersonite	$Na_2Ca(UO_2)(CO_3)_3 \cdot 6H_2O$	39.2	
Autunite	$Ca(UO_2)_2(PO_4)_2 \cdot 10-12H_2O$	45.4-48.2	
Calciumphosphouranite.....	Synonym		
Calcouranite.....	Synonym		
Lime-uranite.....	Synonym		
Sodium autunite.....	$Na(UO_2)_2(PO_4) \cdot 8H_2O$; variety of autunite	52.0	
Bassetite	$Fe(UO_2)_2(PO_4)_2 \cdot 8H_2O$	51.0	
Bayleyite	$Mg_2(UO_2)(CO_3)_3 \cdot 18H_2O$	28.9	
Becquerelite	$CaO \cdot 6UO_3 \cdot 11H_2O$	72.5	
Betafite	$(U, Ca)(Nb, Ta, Ti)_3O_9 \cdot nH_2O?$	13.7-24.5	1.0-1.1
Blomstrandite.....	Composition uncertain		
Mendeleyevite.....	Synonym	16.3	
	Titanian betafite	13.7	
Samiresite.....	Has been considered a plumbian variety of betafite	18.7	
Beta-uranophane	$Ca(UO_2)_2(SiO_3)_2(OH)_2 \cdot 5H_2O$	55.6	
Beta-uranotile.....	Synonym		
Randite.....	A mixture of beta-uranophane, some tyuyamunite, and calcite		
Billietite	$BaO \cdot 6UO_3 \cdot 11H_2O$	68.8	
Boltwoodite	$K_2(UO_2)_2(SiO_3)_2(OH)_2 \cdot 5H_2O$	48.8	
Brannerite	$(U, Ca, Fe, Y, Th)_3(Ti, Si)_5O_{16}?$	27.9-43.6	0.26-4.4
Absite.....	Thorian brannerite; possibly $2UO_3 \cdot ThO_2 \cdot 7TiO_2 \cdot 5H_2O$	26.5	11.3
Lodochnikite.....	Variety of brannerite; possibly $2(U, Th)O_2 \cdot 3UO_2 \cdot 14TiO_2$	44.0	3.5
Calciosamarskite	Probably $(Ca, X, U, Th)_3(Nb, Ta, Fe, Ti, Sn)_5O_{16}(?)$ X = yttrium and other rare earths	9.4-11.3	1.9-2.9
Carnotite	$K_2(UO_2)_2(VO_4)_2 \cdot 3H_2O$; H_2O can range 1 to 3	52.8-55.0	
Cheralite	Isostructural with monazite, AXO_4 , with A = Th, Ca, Ce, La, U, Pb; X = P, Si	3.5-5.5	25.9-27.7
Clarkeite	$(Na, K)_{2-2x}(Ca, Pb)_xU_2O_7 \cdot H_2O$	68.0-68.9	
Coffinite	$U(SiO_4)_{1-x}(OH)_{4x}$ Most specimens associated with carbonaceous material	60.2 (in concentrated, but not pure sample)	

Table 6-9. Uranium and Thorium Minerals (Continued)

Name	Nominal composition: comments	Nominal, typical, or average content or range	
		U, %	Th, %
Nenadkevite.....	Probably a variety of coffinite containing considerable Ca, Mg, Pb		
Cuprosklodowskite.....	$\text{Cu}(\text{UO}_2)_2(\text{SiO}_3)_2(\text{OH})_2 \cdot 5\text{H}_2\text{O}$	54.1	
Jachymovite.....	Synonym		
Kieselkupferuranoxyd..	Synonym		
Cuprozippeite.....	$\text{Cu}(\text{UO}_2)_3(\text{SO}_4)_3(\text{OH})_2 \cdot 11\text{H}_2\text{O}$. Validity questionable		
Curite.....	$3\text{PbO} \cdot 8\text{UO}_3 \cdot 4\text{H}_2\text{O}$	62.9	
Davidite.....	$\text{AB}_3(\text{O},\text{OH})_7$ or AB_3O_7 A = Fe^2 , rare earths, U^6 , Ca, Zr, Th; B = Ti, Fe^3 , V, Cr	≤ 4.4	≤ 0.12
Ferutite.....	Variety of davidite	2.7-9.8	0.06-0.12
Mavudzite.....	Synonym of ferutite		
Delorenzite.....	$(\text{Y},\text{U},\text{Fe}^2)(\text{Ti},\text{Sn})_3\text{O}_8?$	8.7	
Dewindtite.....	$\text{Pb}_3(\text{UO}_2)_5(\text{PO}_4)_4 \cdot 10\text{H}_2\text{O}?$	49.5	
Stasite.....	Synonym		
Drögmansite.....	May be related to sklodowskite		
Dumonite.....	$\text{Pb}_2(\text{UO}_2)_3(\text{PO}_4)_2(\text{OH})_4 \cdot 3\text{H}_2\text{O}$	46.5	
Epanthinite.....	Possibly one of $\text{UO}_2 \cdot 2\text{H}_2\text{O}$ poly- morphs. A rare alteration product of ianthinite	87.8	
Eschynite.....	$(\text{Ce},\text{Ca},\text{Fe}^2,\text{Th})(\text{Ti},\text{Nb})_2\text{O}_6$		9.9-15.4
Euxenite.....	$(\text{Y},\text{Ca},\text{Ce},\text{U},\text{Th})(\text{Nb},\text{Ta},\text{Ti})_2\text{O}_6$	0.6-9.0	≤ 4.3
Eschwegite.....	Tantalian variety of euxenite		
Lyndochite.....	Variety of euxenite-polycrase, relatively high in Ca and Th and low in U		
Oliveiraite.....	Alteration product of euxenite		
Tanteuxenite.....	Variety of euxenite with Ta sub- stituting for Nb		
Titanoniobite.....	Allied to euxenite		
Ferghanite.....	Supposedly $\text{U}_3(\text{VO}_4)_2 \cdot 6\text{H}_2\text{O}$. Perhaps leached or weathered tyuyamunite	67.9	
Fergusonite.....	$(\text{Y},\text{Er},\text{Ce},\text{Fe})(\text{Nb},\text{Ta},\text{Ti})\text{O}_4$	0.8-7.2	0.7-2.5
Adelpholite.....	Synonym of fergusonite? A poorly defined substance, pos- sibly an altered mossite		
Arrhenite.....	An altered fergusonite		
Bragite.....	Synonym		
Kochelite.....	Synonym		
Risorite.....	Synonym		
Rutherfordite.....	An altered fergusonite		
Sipylite.....	Synonym		

Table 6-9. Uranium and Thorium Minerals (Continued)

Name	Nominal composition; comments	Nominal, typical, or average content or range	
		U, %	Th, %
Tyrite	Synonym		
Formanite	(Y,U,Th,Ca)(Ta,Nb,Ti)O ₄	1.1	1.1
Fourmarierite	PbO·4UO ₃ ·7H ₂ O or PbO· 4UO ₃ ·8H ₂ O	63.7 or 63.0	
Francevillite	(Ba,Pb)(UO ₂) ₂ (VO ₄) ₂ ·5H ₂ O	46.0	
Fritzcheite	Mn(UO ₂) ₂ [(P,V)O ₄] ₂ ·8H ₂ O?		
Gastunite	Main components are U, Ca, Si		
Gummite	Generic term for minerals occurring as alteration products of uraninite and not otherwise identified. Group includes silicates, phosphates, and oxides		
Eliasite	Synonym	57.2	
Hydronasturan	Oxidation product of pitchblende		
Parapitchblende	Synonym (?)		
Pittininite	Synonym		
Yitrogummite	Yttrian variety of gummite. An alteration product of yttrian uraninite		
Huttonite	ThSiO ₄		71.6
Ianthinite	2UO ₂ ·7H ₂ O?	71.5	
Iriginite	UO ₃ ·2MoO ₃ ·4H ₂ O	34.9	
Priguinite	Synonym		
Irinite	(Na,Ce,Th) _{1-x} (Ti,Nb)-[O _{3-x} (OH) _x]		11.4
Ishikawaite	(U,Fe,X)(Nb,Ta)O ₄ . X = rare earths	19.3	
Johannite	Cu(UO ₂) ₂ (SO ₄) ₂ (OH) ₂ ·6H ₂ O	50.8	
Gilpinite	Synonym		
Peligoite	Probably identical with johannite		
Uranvitriol	Synonym		
Kahlerite	Fe(UO ₂) ₂ (AsO ₄) ₂ ·nH ₂ O	46.8(?)	
Kasolite	Pb(UO ₂)(SiO ₃)(OH) ₂	40.5	
Khlopinit	(Y,U ⁴ ,Th) ₃ (Nb,Ta,Ti,Fe) ₇ . O ₂₀ ?	7.2	1.9
Lermontovite	(U,Ca,X) ₃ (PO ₄) ₄ ·6H ₂ O	44.3	
Liebigite	Ca ₂ (UO ₂)(CO ₃) ₃ ·10-11H ₂ O	33.6	
Flutherite	Synonym		
Kalk-uran-carbonat	Synonym		
Uranothallite	Synonym		
Masuyite	UO ₃ ·2H ₂ O	73.9	
Medjidite	Supposedly a uranium sulfate. Validity of species is questionable		

Table 6-9. Uranium and Thorium Minerals (Continued)

Name	Nominal composition; comments	Nominal, typical, or average content or range	
		U, %	Th, %
Meta-autunite I	$\text{Ca}(\text{UO}_2)_2(\text{PO}_4)_2 \cdot 2\frac{1}{2}\text{--}6\frac{1}{2}\text{H}_2\text{O}$. Apparently not formed directly in nature, but most field and museum specimens of autunite have been dehy- drated to this phase	52.4–57.1	
Metakalkuranite	Synonym of meta-autunite		
Meta-autunite II	$\text{Ca}(\text{UO}_2)_2(\text{PO}_4)_2 \cdot \text{O--}6\text{H}_2\text{O}$. Not found in nature. Meta-autu- nite I passes into this phase on heating to about 80°C	53.0–60.4	
Metatorbernite	$\text{Cu}(\text{UO}_2)_2(\text{PO}_4)_2 \cdot n\text{H}_2\text{O}$; $n = 4$ to 8	50.8	
Metachalcolite	Synonym		
Metakupferuranit	Synonym		
Metatyuyamunite	$\text{Ca}(\text{UO}_2)_2(\text{VO}_4)_2 \cdot 5\text{--}7\text{H}_2\text{O}$	50.1–52.2	
Meta-uranocircite	$\text{Ba}(\text{UO}_2)_2(\text{PO}_4)_2 \cdot 8\text{H}_2\text{O}$	47.1	
Bariumphosphouranit	Synonym		
Uranocircite	Synonym		
Meta-uranopilite	$(\text{UO}_2)_6(\text{SO}_4)(\text{OH})_{10} \cdot 5\text{H}_2\text{O}$? Validity of species is doubtful	72.3 (?)	
Beta-uranopilite	Synonym		
Metazeunerite	$\text{Cu}(\text{UO}_2)_2(\text{AsO}_4)_2 \cdot 8\text{H}_2\text{O}$	46.4	
Kupferarsenuranit	Synonym of zeunerite		
Zeunerite	All natural specimens labeled zeunerite have proved to be metazeunerite. The fully hy- drated, or true, zeunerite has not yet been found in nature		
Moluranite	$\text{UO}_2 \cdot 2\text{UO}_3 \cdot 5\text{MoO}_3 \cdot 12\text{H}_2\text{O}$	36(?)	
Monazite	$(\text{Ce, La, Th})\text{PO}_4$.		Normally from a few % to 10.6%, but series probably extends to 26.4%
Cryptolite			
Edwardsite			
Eremite			
Guadarramite	An intergrowth of radioactive monazite and some ilmenite		
Kararfveite	Impure monazite		
Mengite	Synonym		
Monazitoid	Synonym		

Table 6-9. Uranium and Thorium Minerals (Continued)

Name	Nominal composition; comments	Nominal, typical, or average content or range	
		U, %	Th, %
Phosphocerite.....	Synonym		
Turnerite.....	Synonym		
Urdite.....	Synonym		
Nohelite	$(\text{Ca}, \text{Mg}, \text{Fe}^2, \text{X}, \text{U})_2(\text{Nb}, \text{Zr}, \text{Fe}^3)_{3-10}$. X = yttrium and other rare earths. Validity of species is doubtful	13.0	
Novacekite	$\text{Mg}(\text{UO}_2)_2(\text{AsO}_4)_2 \cdot 8-10\text{H}_2\text{O}$	51.6-53.7	
Orlite	$\text{Pb}_3(\text{UO}_2)_3(\text{Si}_2\text{O}_7)_2 \cdot 6\text{H}_2\text{O}$	36.3	
Parsonsite	$\text{Pb}_2(\text{UO}_2)(\text{PO}_4)_2 \cdot \text{H}_2\text{O}$ or $2\text{H}_2\text{O}$	26.1-26.7	
Phosphuranylite	$\text{Ca}(\text{UO}_2)_4(\text{PO}_4)_2(\text{OH})_4 \cdot 7\text{H}_2\text{O}$	63.3	
Pilbarite	$\text{PbO} \cdot \text{UO}_3 \cdot \text{ThO}_2 \cdot 2\text{SiO}_2 \cdot 4\text{H}_2\text{O}$	24.4	27.4
Pisekite	A niobate-tantalate-titanate of U and rare earths with Th and Sn		
Polycrase	$(\text{Y}, \text{Ca}, \text{Ce}, \text{U}, \text{Th})(\text{Ti}, \text{Nb}, \text{Ta})_2\text{O}_6$	5.5-12.4	<4.7
Priorite	$(\text{Y}, \text{Er}, \text{Ca}, \text{Fe}^2, \text{Th})(\text{Ti}, \text{Nb})_2\text{O}_6$	0.4-3.4	0.5-14.9
Blomstrandine.....			
Rabbittite	$\text{Ca}_3\text{Mg}_3(\text{UO}_2)_2(\text{CO}_3)_6(\text{OH})_4 \cdot 18\text{H}_2\text{O}$	31.1	
Rauvite	$\text{CaO} \cdot 2\text{UO}_3 \cdot 5\text{V}_2\text{O}_5 \cdot 16\text{H}_2\text{O}?$	26.1	
Renardite	$\text{Pb}(\text{UO}_2)_4(\text{PO}_4)_2(\text{OH})_4 \cdot 7\text{H}_2\text{O}$	57.3	
Richetite	Contains Pb and U. Validity of species is questionable		
Rutherfordine	UO_2CO_3	72.1	
Diderichite.....			
Sabugalite	$\text{HA}1(\text{UO}_2)_4(\text{PO}_4)_4 \cdot 16\text{H}_2\text{O}$	53.6	
Saleeite	$\text{Mg}(\text{UO}_2)_2(\text{PO}_4)_2 \cdot 8-10\text{H}_2\text{O}$	50.9-53.0	
Magnesium-phosphouranite.....			
Samarskite	$(\text{Y}, \text{Ce}, \text{U}, \text{Ca}, \text{Fe}, \text{Pb}, \text{Th})(\text{Nb}, \text{Ta}, \text{Ti}, \text{Sn})_2\text{O}_6$	8.4-16.1	<3.7
Annerodite.....	Samarskite with parallel overgrowths of columbite		
Eytlandite.....			
Hydrosamarskite.....	An altered samarskite		
Nuevite.....			
Plumboniobite.....	A niobate of Y, U, Pb, Fe and rare earths. May be a plumbian variety of samarskite		
Rogersite.....	Probably an altered samarskite; of doubtful validity		
Uranotantalite.....			
Vietinghofite.....	Supposedly a ferroan samarskite		
Yttro-ilmenite.....			

Table 6-9. Uranium and Thorium Minerals (Continued)

Name	Nominal composition; comments	Nominal, typical, or average content or range	
		U, %	Th, %
Schoepite	Close to $\text{UO}_3 \cdot 2\text{H}_2\text{O}$ but perhaps $8\text{UO}_3 \cdot 2\text{OH}_2\text{O}$	71.9	
Paraschoepite.....	Identical with schoepite		
Schroëckingerite	$\text{NaCa}_3(\text{UO}_2)(\text{CO}_3)_3(\text{SO}_4)\text{F} \cdot$ $10\text{H}_2\text{O}$	26.8	
Dakeite.....			
Neogastunite.....			
Sengierite	$\text{Cu}(\text{UO}_2)(\text{VO}_4)(\text{OH}) \cdot 4\text{--}5\text{H}_2\text{O}?$	42.8–44.3	
Sharpite	$(\text{UO}_2)(\text{CO}_3) \cdot \text{H}_2\text{O}$ or $(\text{UO}_2)_6(\text{CO}_3)_5(\text{OH})_2 \cdot 7\text{H}_2\text{O}$	68.4 or 68.6	
Skłodowskite	$\text{Mg}(\text{UO}_2)_2(\text{SiO}_3)_2(\text{OH})_2 \cdot 6\text{H}_2\text{O}$	55.6	
Chinkolobwite.....			
Soddyite	$(\text{UO}_2)_5(\text{SiO}_4)_2(\text{OH})_2 \cdot 5\text{H}_2\text{O}$	71.8	
Studdite	Hydrated carbonate of U and Pb. Species not well-defined		
Swartzite	$\text{CaMg}(\text{UO}_2)(\text{CO}_3)_3 \cdot 12\text{H}_2\text{O}$	32.6	
Thorianite	$(\text{Th}, \text{U})\text{O}_2$	<46.5	45.3–87.9
Aldanite.....	A variety of thorianite	12.4–24.1	
Uranothorite.....	Synonym for thorianite contain- ing U		
Thorite	ThSiO_4	<10.1	25.2–62.7
Auerlite.....	Phosphatian variety of thorite		
Calciorthorite.....	Variety of thorite		
Enalite.....	Uranoan variety of thorite		
Eucrasite.....	Possibly a variety of thorite high in rare earths		
Ferrothorite.....	A ferroan variety of thorite		
Freyalite.....	Variety of thorite high in rare earths		
Orangite.....			
Uranothorite.....	Uranoan variety of thorite		
Wisaksonite.....	Uranoan variety of thorite		
Thorogummite	$\text{Th}(\text{SiO}_4)_{1-x}(\text{OH})_{4x}$	2.5–31.4	18.2–50.8
Chlorothorite.....			
Hyblite.....			
Hydrothorite.....			
Mackinioshite.....			
Maitlandite.....			
Nicolayite.....			
Thucholite	A complex of uraninite with hy- drocarbons	<53 in the ash	<48 in the ash
Carbocer.....	A mineral similar to thucholite, high in Ce		
Carburan.....	A hydrocarbon complex related to thucholite	4.29	

Table 6-9. Uranium and Thorium Minerals (Continued)

Name	Nominal composition; comments	Nominal, typical, or average content or range	
		U, %	Th, %
Titanothucholite.....	Titanian variety of thucholite	47.1	
Torbernite	$\text{Cu}(\text{UO}_2)_2(\text{PO}_4)_2 \cdot 12\text{H}_2\text{O}$		
Chalkolith, chalcolite...			
Copper uranite.....			
Cuprouranit.....			
Kupferphosphouranit..			
Torberite.....			
Uranite.....	Synonym of torbernite-autunite group	55.1	
Uran-mica.....			
Uranphullit.....			
Troegerite	Probably $\text{H}_2(\text{UO}_2)_2(\text{AsO}_4)_2 \cdot 8\text{H}_2\text{O}$	55.1	<18.4
Tscheffkinit	Complex silicate of rare earths, Fe, Mn, Mg, Ca, Al, Ti, Th, and U	2.3	
Perrierite.....	Perhaps identical with tscheffki- nite		
Tyuyamunite	$\text{Ca}(\text{UO}_2)_2(\text{VO}_4)_2 \cdot 7\text{--}10.5\text{H}_2\text{O}$	54.4-56.7	<45.3
Calciocarnotite.....			
Umohoite	Close to $(\text{UO}_2)(\text{MoO}_4) \cdot 4\text{H}_2\text{O}$	47.4	
Unnamed mineral (Antipov, 1900)	Essentially $\text{Cu}(\text{UO}_2)(\text{CO}_3)_2 \cdot 10\text{H}_2\text{O}$	37.6	
Unnamed mineral (Chernik, 1922)	Ill-defined vanadate of Cu and U	32.5	
Unnamed mineral (Chirvinsky, 1925)	May be variety of tyuyamunite		
Uraconite	This name has been used for uranium sulfates but lacks specific meaning		
Calciouraconite.....	Near $\text{Ca}(\text{UO}_2)_4(\text{SO}_4)_2(\text{OH})_6 \cdot 20\text{H}_2\text{O}$		
Uraninite	Ideally UO_2 . Usually more or less oxidized and ranging in composition to at least $(\text{U}^{4+}, \text{U}^{6+})\text{O}_{2.6}$. Also contains Th, Pb, Y, and other rare earths, in solid solution. Forms com- plete series with thorianite. Division between uraninite and thorianite at $\text{U}:\text{Th} = 1:1$	46.5-88.2	
Broggerite.....	A thorian variety of uraninite		
Caracite.....			
Cleveite.....	A variety of uraninite contain- ing rare earths		

Table 6-9. Uranium and Thorium Minerals (Continued)

Name	Nominal composition; comments	Nominal, typical, or average content or range	
		U, %	Th, %
Nasturan.....	Synonym of pitchblende		
Nivenite.....	A variety of uraninite contain- ing rare earths		
Pitchblende.....	A colloform, fine-grained variety of uraninite		
Ulrichite.....			
Uranoniobite.....			
Uranopissite.....			
Uranochalcite.....	An ill-defined uranium sulfate of doubtful validity		
Uranocher.....	A generic term used chiefly for uranium sulfates and, in part, uranium oxides		
Uranophane.....	$\text{Ca}(\text{UO}_2)_2(\text{SiO}_3)_2(\text{OH})_2 \cdot 5\text{H}_2\text{O}$	55.6	
Lambertite.....			
Uranotil, uranotile.....			
Uranopilite.....	$(\text{UO}_2)_6(\text{SO}_4)(\text{OH})_{10} \cdot 12\text{H}_2\text{O}$	67.9	
Uranospathite.....	$\text{Cu}(\text{UO}_2)_2(\text{AsO}_4)_2 \cdot 11\text{H}_2\text{O}?$	46.0(?)	
Uranosphaerite.....	$(\text{Bi})(\text{UO}_2)(\text{OH})_3?$	43.6(?)	
Uranospinite.....	$\text{Ca}(\text{UO}_2)_2(\text{AsO}_4)_2 \cdot 10\text{H}_2\text{O}$	45.9	
Calciumarsenuranit.....			
Uvanite.....	$\text{U}_2\text{V}_6\text{O}_{21} \cdot 15\text{H}_2?$	34.3(?)	
Vandenbrandeite.....	$\text{Cu}(\text{UO}_2)_2\text{O}_2 \cdot 2\text{H}_2\text{O}$	59.3	
Uranolepidite.....			
Vandendriesscheite.....	$\text{PbO} \cdot 7\text{UO}_3 \cdot 12\text{H}_2\text{O}$	68.3	
Voglianite.....	A hydrous calcium and uranium sulfate of doubtful validity		
Voglite.....	$\text{Ca}_2\text{CuU}(\text{CO}_3)_6 \cdot 6\text{H}_2\text{O}(?)$	32.6	
Uran-kalk- kupfer carbonat.....			
Walpurgite.....	Probably $\text{Bi}_4(\text{UO}_2)(\text{AsO}_4) \cdot$ $3\text{H}_2\text{O}$	15.9	
Zippeite.....	$3\text{UO}_3 \cdot 2\text{SO}_3 \cdot 9\text{H}_2\text{O}?$ or $2\text{UO}_3 \cdot \text{SO}_3 \cdot 5\text{H}_2\text{O}?$	59.1-64.1	

Table 6-10. Minerals with Minor Amounts of Uranium and Thorium

Name	Nominal composition; comments	Nominal, typical, or average content or range	
		U, %	Th, %
Abukumalite	(Th,Ca,Y) ₅ (SiO ₄ ,AlO ₄) ₃ (O,F)		0.8
Allanite	(Ca,Ce,Th) ₂ (Al,Fe,Mg) ₃ Si ₃ O ₁₂ (OH)	0.02	≤3.2
Bagrationite.....	Variety of allanite. Name also refers to cerian variety of epidote		
Bodenite.....	Related to muromontite in composition		
Bucklandite.....			
Muromontite.....	Apparently related to allanite but containing yttrium and beryllium		
Nagetelite.....	Phosphatian variety of allanite		
Orthite.....			
Uralorthite.....			
Wasite.....	Altered allanite		
Xanthorite.....	Altered allanite		
Yttro-orthite.....	Variety of orthite containing 8% Y ₂ O ₃		
Anthraxolite	A nickeliferous and uraniferous hydrocarbon	0.003	
Asphaltite	Includes solid bituminous hydrocarbons known as albertite, impsonite, gilsonite, grahamite, nigrite, and uintaite	0.001	
Broggite.....	A variety of asphaltite		
Bastnaesite	(Ce,La)FCO ₃	<1	<1
Buszite.....			
Cappelenite	Ba(Y,Ce,La) ₆ B ₆ O ₁₂ (OH) ₂ (SiO ₄) ₃		0.42
Carbonate-fluorapatite ...	A carbonation variety of fluorapatite which is an important constituent of many phosphate rocks	≤0.02	
Cerite	A cerium silicate with minor Ca and Fe	0.4	0.3
Chinglusuite	A complex silicate of Na, Mn, Ca, and Ti, containing small amounts of Th and rare earths		
Columbite	(Fe,Mn)(Nb,Ta) ₂ O ₆	<9.8	
Baierite.....			
Dianite.....			
Ferrocolumbite.....			
Ferro-ilmenite.....			

Table 6-10. Minerals with Minor Amounts of Uranium and Thorium (Continued)

Name	Nominal composition; comments	Nominal, typical, or average content or range	
		U, %	Th, %
Germannolite.....			
Greenlandite.....			
Manganocolumbite....	Variety of columbite		
Toddite.....	Probably a uranoan variety of columbite		
Cordylite.....	$(\text{Ce,La})_2\text{Ba}(\text{CO}_3)_3\text{F}_2$		0.26
Corvusite.....	$\text{V}_2\text{O}_4 \cdot 6\text{V}_2\text{O}_5 \cdot x\text{H}_2\text{O}$	1.45	
Fersmite.....	$(\text{Ca,Ce,Na})(\text{Nb,Ti,Fe,Al})_2(\text{O,OH,F})_6$		0.42
Hielmite.....	AB_2O_6 or $\text{A}_2\text{B}_3\text{O}_{10}$ A = Y, Fe ² , U ⁴ , Mn, Ca	4.0-4.3	
Johnstrupite.....	A complex silicate of Na, Ca, Th, Ce, and Ti		0.7
Kolm.....	Material resembling oil shale	0.43	
Lovchorrite.....	$\text{Ce}_2(\text{TiO}_3)_3 \cdot 10\text{CaSiO}_3 \cdot 2\text{CeF}_3$	0.03	≤0.7
Vudyavrite.....	An altered loychorrite		<1
Lovozerite.....	A complex silicate of Ti and Zr		0.50
Melanocerite.....	Chiefly a borosilicate of the Ce and Y metals		1.5
Caryocerite.....	Near molanocerite, but contains more Th		12.0
Microlite.....	$(\text{Na,Ca})_2(\text{Ta,Nb})_2\text{O}_6(\text{O,OH,F})$	<10.4	0.2
Calciotantalite.....	Possibly a mixture of microlite and tantalite		
Djalmaite.....			
Haddamite.....			
Metasimpsonite.....	An alteration product of simp- sonite; later identified with microlite		
Neotantalite.....	An altered microlite with com- position close to tantalite		
Niobtantal pyrochlore..			
Tantal pyrochlore.....			
Mosandrite.....	Complex silicate of Na, Ca, Ce, and Ti		0.3
Polymignyte.....	$\text{A}(\text{Nb,Ti,Ta})\text{O}_4$ A = Ca, Fe ² , Y, Zr, Th		3.4
Pyrochlore.....	$(\text{Na,Ca})_2(\text{Nb,Ta})_2\text{O}_6\text{F}$	<1.4 usu- ally, but as much as 17.1 in some varieties	<4.4
Azor-pyrrhit.....			

Table 6-10. Minerals with Minor Amounts of Uranium and Thorium (Continued)

Name	Nominal composition; comments	Nominal, typical, or average content or range	
		U, %	Th, %
Chalcolamprite.....	An altered uranoan variety of pyrochlore related to hatchet- tolite		
Ellsworthite.....			
Endeolite.....			
Fluochlore.....	Similar to chalcolamprite in composition. An altered pyrochlore?		
Hatchettolite.....			
Hydrochlore.....			
Koppite.....			
Marignacite.....			
Niobpyrochlor.....			
Pyrrhite.....			
Uran-pyrochlore.....			
Rinkite	$\text{Ce}_2(\text{TiO}_3)_3 \cdot 10\text{CaSiO}_3 \cdot 3\text{CaF}_2$		Small amounts
Rinkolite	Complex silicate of Na, Ca, Ce, and Zr		≤ 0.41
Rowlandite	$(\text{Y, La, Ce})_4\text{Fe}(\text{F, Si}_2\text{O}_7)_2?$	0.4	
Steenstrupine	Complex silicate of rare earths, Th, Na, K, Fe, Mn, Mg, P, Be, Al, and Ta, with (OH) and F		6.2
Tengerite	$\text{CaY}_3(\text{CO}_3)_4(\text{OH})_3 \cdot 3\text{H}_2\text{O}?$		0.26
Thalenite	$\text{Y}_4\text{Si}_4\text{O}_{13}(\text{OH})_2$		0.16
Tritomite	A borosilicate of Ce, Y, Th, Ca, and F		7.5-8.3
Turanite	Supposedly $\text{Cu}_5(\text{VO}_4)_2(\text{OH})_4$	3.2	
Vanoxite	Perhaps $2\text{V}_2\text{O}_4 \cdot \text{V}_2\text{O}_5 \cdot 8\text{H}_2\text{O}$; species may be invalid	≤ 0.5	
Volborthite	Perhaps $\text{Cu}_3(\text{VO}_4)_2 \cdot 3\text{H}_2\text{O}$	3.1	
Calciovolborthite.....	Probably $\text{CuCa}(\text{VO}_4)(\text{HO})$; pos- sibly only a calcian variety of volborthite		
Tangeite.....	Appears to be identical with calciovolborthite		
Uzbekite.....			
Wiikite	Ill-defined mixture and altera- tion product of minerals high in Nb, Ta, Ti, Si, and Y	≤ 13.2	0.09-3.7
Nuolaite.....	A mixture similar to wiikite		1.8-3.5
Xenotime	YPO_4	≤ 3.6	≤ 2.2
Yttrialite	Silicate, chiefly of Th and Y metals	0.8	10.5

Table 6-10. Minerals with Minor Amounts of Uranium and Thorium (Continued)

Name	Nominal composition; comments	Nominal, typical, or average content or range	
		U, %	Th, %
Yttrocrasite	$(Y, Th, U, Ca)_2Ti_4O_{11}?$	2.3	7.7
Yttrotantalite	$(Fe, Y, U)(Nb, Ta)O_4$	≤ 4.8	≤ 1.6
Yttrocolumbite.....	Similar to yttrotantalite		
Zircon	$ZrSiO_4$	≤ 2.7 Usually low	≤ 13.1 ; Usually low
Alvite.....	Variety of zircon, near cyrtolite		
Azorite.....	Variety of zircon		
Calypsolite.....	Probably altered zircon		
Cyrtolite.....	Altered zircon, containing U, Th, Y, and other rare earths		
Hagatalite.....			
Hoegtveitite.....	May be alvite		
Malacon.....	Hydrated or altered zircon con- taining the common elements, but no U, Th, or rare earths except in trace amounts		
Naegite.....	Variety of zircon		
Oerstedite.....	Altered zircon		
Oyamalite.....	Variety of zircon		
Ribeirite.....	An altered zircon		
Tachyaphaltite.....	Probably altered zircon		
Yamagutilite.....	Variety of zircon; contains P_2O_5 and rare earths		
Zirkelite	$(Ca, Fe, Th, U)_2(Ti, Zr)_2O_5?$	1.4	6.4

Table 6-11. Potassium Minerals

Name	Nominal composition; comments	K content, %
Evaporites and other water-soluble salts		
Sylvine	KCl	nom. 52.4
Sylvite.....		
Natrikalite.....	(Na,K)Cl. Almost certainly a mixture of sylvine and halite (NaCl)	~10-20
Sylvinite.....	Commercial name for sylvine or natrikalite	
Hard salt.....	Impure sylvite containing halite (NaCl) and kieserite ($\text{MgSO}_4 \cdot \text{H}_2\text{O}$)	~10-14
Carnallite	$\text{KMgCl}_3 \cdot 6\text{H}_2\text{O}$	nom. 14.1
Carnallite rock.....	Impure carnallite	~8-10
Niter	KNO_3	nom. 38.7
Saltpeter.....		
Nitrokalite.....		
Kalite.....		
Kalicine	KHCO_3	nom. 39.1
Aphthitalite	$(\text{Na,K})_2\text{SO}_4 \cdot \frac{1}{3} \lesssim \text{Na}:\text{K} \lesssim 3$	~23-35
Glaserite.....	Most common variety of aphthitalite, with $\text{Na}:\text{K} \approx \frac{1}{3}$	~35
Arcanite.....	Hypothetical end member of aphthitalite series with no Na; hence K_2SO_4	(nom. 44.3)
Mercallite	KHSO_4	nom. 28.7
Langbeinite	$\text{K}_2\text{Mg}_2(\text{SO}_4)_3$	nom. 18.8
Leonite	$\text{K}_2\text{Mg}(\text{SO}_4)_2 \cdot 4\text{H}_2\text{O}$	nom. 21.3
Kalium-astrakanite.....		
Kalium-blödite.....		
Picromerite	$\text{K}_2\text{Mg}(\text{SO}_4)_2 \cdot 6\text{H}_2\text{O}$	nom. 19.4
Schönite.....		
Kainite	$\text{KMgSO}_4\text{Cl} \cdot 3\text{H}_2\text{O}$	nom. 15.7
Anhydrokainite.....	Dehydrated kainite	nom. 20.0
Syngenite	$\text{K}_2\text{CaSO}_4 \cdot \text{H}_2\text{O}$	nom. 31.0
Polyhalite	$\text{K}_2\text{MgCa}_2(\text{SO}_4)_4 \cdot 2\text{H}_2\text{O}$	nom. 13.0
Alum	$\text{KAl}(\text{SO}_4)_2 \cdot 12\text{H}_2\text{O}$	nom. 8.2
Silicates occurring principally in igneous and metamorphic rocks		
Kaliophilite	KAlSiO_4	nom. 24.7
Kalsilite	KAlSiO_4	nom. 24.7
Nepheline	NaAlSiO_4 , with K replacing some Na	<7
Potash-nepheline.....	Variety of nepheline close to $\text{Na}_3\text{KAl}_4\text{Si}_4\text{O}_{16}$	~7
Leucite	KAlSi_2O_6 ; some K may be replaced by Na	≤17.9
Amphigene.....		
Microcline	KAlSi_3O_8	nom. 15.1

Table 6-11. Potassium Minerals (*Continued*)

Name	Nominal composition; comments	K content, %
Silicates occurring principally in igneous and metamorphic rocks (<i>Continued</i>)		
Orthoclase	KAlSi_3O_8	nom. 15.1
Potassium feldspar		
Common feldspar		
Adularia	Variety of orthoclase	
Loxoclase	Variety of orthoclase with Na replacing some K	<15
Soda-orthoclase		
Perthite	Lamellar intergrowth of plagioclase, $(\text{NaSi}, \text{CaAl})\text{AlSi}_2\text{O}_8$, in orthoclase	
Antiperthite	Lamellar intergrowth of orthoclase in plagioclase	
Sanidine	$(\text{K}, \text{Na})\text{AlSi}_3\text{O}_8$, low in Na	<7
Glassy feldspar		
Natronsandine	Variety of sanidine relatively high in Na	<<7
Anorthoclase	$(\text{Na}, \text{K})\text{AlSi}_3\text{O}_8$	<10
Soda-microcline		
Potash-anorthoclase	Variety of anorthoclase with $\text{K}:\text{Na} \approx 2$	~10
Potash-albite	Variety of albite, $\text{NaAlSi}_3\text{O}_8$, containing K in place of some Na	Several
Potash-anorthite	Variety of anorthite, $\text{CaAl}_2\text{Si}_2\text{O}_8$, with KSi in place of some CaAl; $\text{K}:\text{Ca} > 0.1$	>1.3
Potash-bytownite	Variety of bytownite, $(\text{NaSi}, \text{CaAl})\text{AlSi}_2\text{O}_8$, containing appreciable amounts of K in place of Na	Several
Potash-labradorite	Variety of labradorite, $(\text{NaSi}, \text{CaAl})\text{-AlSi}_2\text{O}_8$, containing appreciable amounts of K in place of Na	Several
Potash-andesine	Variety of andesine, $(\text{NaSi}, \text{CaAl})\text{AlSi}_2\text{O}_8$, containing appreciable amounts of K in place of Na	Several
Potash-oligoclase	Variety of oligoclase, $(\text{NaSi}, \text{CaAl})\text{AlSi}_2\text{O}_8$, containing appreciable amounts of K in place of Na	Several
Muscovite	$\text{KAl}_3\text{Si}_3\text{O}_{10}(\text{OH})_2$	nom. 9.8
White mica		avg. 8.6
Potash mica		
Hydromuscovite	Variety of muscovite with higher OH and lower K	<9
Lithium muscovite	Variety of muscovite with Li in place of some Al	~10
Phengite	Variety of muscovite	
Sericite	Variety of muscovite	
Biotite	$\text{K}_2(\text{Mg}, \text{Fe}^{\text{ii}}, \text{Al}, \text{Fe}^{\text{iii}})_{4-6}(\text{Si}, \text{Al})_8\text{O}_{20}(\text{OH})_4$	avg. 6.8
Black mica		

Table 6-11. Potassium Minerals (*Continued*)

Name	Nominal composition; comments	K content, %
Silicates occurring principally in igneous and metamorphic rocks (<i>Continued</i>)		
Iron mica.....	$K_2(Mg, Fe^{ii}, Fe^{iii})_{4-6} (Al, Fe^{iii}, Si)_8 O_{20}(OH)_4$	~8
Magnesia mica.....		
Hydroxyl biotite.....		
Lepidomelane		
Ferribiotite.....		
Ferromuscovite.....	$KMg_3AlSi_3O_{10}(OH)_2$	nom. 9.4
Iron mica.....		
Phlogopite		
Magnesium mica.....	$K(Li, Al)_3(Si, Al)_4O_{10}(F, OH)_2$	~10
Lepidolite		
Lithium mica.....		
Hornblende	$(Na, K, Ca)_{2-3}(Mg, Fe^{ii}, Al, Fe^{iii})_5 - (Si, Al)_8 O_{22}(OH)_2$	Several
Silicates occurring principally in sedimentary rocks, clays, and soils		
Illite	$K_{2-3}Al_1Si_{12-13}O_{36-38}(OH)_{12-13}$. Also, group name for micaceous clays	5-8
Montmorillonite	Near $RMgAl_5Si_{12}O_{30}(OH)_8 \cdot nH_2O$; R = Na, K, $\frac{1}{2}Ca$ present as exchangeable bases; composition variable	<3
Fuller's earth.....	An altered volcanic rock consisting largely of montmorillonite	≤3.8
Bentonite.....		
Glacialite.....		
Potash-montmorillonite.....	Variety of montmorillonite high in K	~7
Glaucanite	Near $K_3(Mg, Fe^{ii}, Al, Fe^{iii})_8 -_{12} (Si, Al)_{16} O_{40}(OH)_8$	
Green earth.....	Variety of glauconite with about half of the K replaced by Na	~4
Soda-glaucanite.....		
Hydromica	Group name for micas with much water	Always contain some K
Zeolite	Family name for hydrated aluminosilicates of alkali and alkaline earth elements; composition variable	Usually contain some K

of potassium, the content in a rock as a whole is generally closely related to the composition of the essential (most abundant) mineral constituents. On the other hand, the bulk of the uranium and of the thorium in rocks is often present in accessory (minor) minerals or in amorphous, fluid, or indeterminate occlusions or interstitial deposits. Thus, although the *potassium* content of rocks is usually less than 5 per cent and only rarely exceeds 10 per cent, the *uranium* and *thorium* contents can be and often are considerably in excess of the average or typical values; in such cases their contributions to the radioactivity are much greater than that of potassium. So, although most igneous rocks have less than ~ 7 ppm U and ~ 25 ppm Th, occasional samples having up to several hundred parts per million of these two elements exist. Sedimentary rock types can show even higher occasional concentrations: marine phosphate rocks having over 0.1 per cent U and beach placers having several tenths of a per cent of Th are known.

Among the **igneous rocks** of the earth's crust, K, U, and Th show roughly parallel variations with each other and with acidity (silica content), which in turn is correlated with density and consequently with average depth of occurrence. The most acidic rocks, which are lightest and occur close to the surface, have the highest radioelement concentrations, whereas the most basic and heaviest rocks, which occur mainly at great depth but are found occasionally at the surface, have much lower contents of these elements. **Potassium** ranges from ~ 2 to ~ 6 per cent and averages ~ 3 per cent in granites; obsidians and other acidic rocks have similar contents. Intermediate-type rocks range from ~ 1 to ~ 3 per cent K. Typical of basic rocks, basalts range from less than 0.2 to over 2 per cent, averaging ~ 0.5 per cent K. Dunites average ~ 0.001 per cent K and are probably representative of the ultrabasic rocks in general. **Uranium** is typically ~ 6 ppm in obsidians, ~ 3 to 5 ppm in granites, ~ 2 to 3 ppm in intermediate rock types, ~ 1 ppm in basic rocks, and ~ 0.05 ppm in ultrabasics. The igneous rocks highest in uranium are certain alkali-rich granites, in which uranium contents may rise to over 100 ppm. Thorium parallels uranium fairly closely, the Th/U ratio being ~ 3 to 4 in most igneous rocks. In alkali-rich granites the Th content may occasionally exceed 300 ppm.

In **sedimentary rocks** and their metamorphic derivatives the over-all content of potassium is ~ 2.4 per cent. Shales and other argillaceous (clayey) sediments average ~ 3 per cent, sandstones and other arenaceous (sandy) sediments ~ 1 per cent, and limestones and other carbonate rocks ~ 0.3 per cent K. Uranium is low in the most abundant sedimentary rocks, averaging slightly over ~ 1 ppm in most siliceous and carbonate types. However, there are many less abundant sedimentary rocks with much higher uranium contents. This is particularly true of carbonaceous types: thus, black shales and schists may have from ~ 5 to ~ 350 ppm U; kolm-bearing shales have ~ 200 ppm U and kolm itself sometimes exceeds 1 per cent U; lignite may have up to ~ 200 ppm U; coal, though usually low in U, commonly has ~ 30 ppm and occasionally as high as 500 ppm U. Phosphate rocks of marine origin are also relatively high in U, commonly having ~ 50 to ~ 200 ppm and occasionally $\sim 1,000$ ppm. Placers (alluvial deposits of heavy sands) average ~ 2 ppm U but may have up to ~ 70 ppm. The thorium content and Th/U ratio are also more variable in sedimentary than in igneous rocks. Argillaceous rocks average ~ 11 ppm, arenaceous rocks ~ 5 ppm, and carbonate rocks ~ 1 ppm Th. Gneiss and

schist may sometimes have up to ~ 100 ppm Th. Thorium is particularly enriched in placer deposits, which average ~ 60 ppm and may have up to $\sim 3,000$ ppm Th. Some other materials of sedimentary derivation are occasionally enriched in radioelements, particularly in uranium. Petroleum may have up to ~ 100 ppm U, and some asphaltites having over 1 per cent U are known. Oil-field brines may have several parts per million of uranium.

The distinction between rocks having relatively high concentrations of a particular element and ores of that element is based on economic factors. In the case of uranium in the United States, moderately extensive deposits which contain over 0.1 per cent U_3O_8 (850 ppm U), which is the lower limit to uranium-containing material that will be purchased by the U.S. Atomic Energy Commission, are regarded as ore deposits. Ores are frequently mixtures of one or more minerals of the element with rocks of common types; so there is a continuous gradation in element content between the lowest values found in rocks to the highest values found in minerals.

More extensive information on the content of rocks and other natural materials in potassium, uranium, and thorium can be found in "Geochemistry" (Rankama, K., and T. G. Sahama, University of Chicago Press, Chicago, 1950); "Nuclear Geology" [Faul, H. (ed.), Wiley, New York, 1954]; and *Geology of Uranium and Thorium* (*Proc. Intern. Conf. Peaceful Uses Atomic Energy, Geneva*, vol. 6, 1955).

There are several types of **secondary natural radionuclide occurrences**. **Hokuto-lite** (synonym: **anglesobarite**) is a rather strongly radioactive sinter deposit formed in hot springs; it is chemically mainly $(Ba,Pb)SO_4$; mineralogically it resembles barite; its radioactivity is due to a high content of Ra-226, Ra-228, and Pb-210 (with their disintegration products), which are leached from underground rocks by the percolating ground waters and deposited preferentially. The activities may be as high as $\sim 10^{-9}$ curie/g. Natural waters, particularly those from hot springs, are often enriched in radioactive lead and barium isotopes. Gases escaping from underground (volcanoes, fumaroles, gas wells, etc.) frequently contain high concentrations of radon; the gas from helium wells may contain from $< 10^{-12}$ to $\sim 10^{-9}$ curie of radon per liter. Measurable amounts of radon and sometimes other emanon isotopes are found in the air as a result of seepage from the soil, but the concentrations are extremely small from the viewpoint of health hazard. In the rivers, seas, and ocean sediments radioactive equilibrium is disturbed; in most deep-sea sediments ionium is enriched by a factor of ~ 30 with respect to uranium, the excess decaying with time and hence with depth in the sediments.

PARTICLE ACCELERATORS

M. Stanley Livingston

ACCELERATOR TYPES

Particle accelerators are of two basic types, **circular** and **straight-line**. Circular machines use magnetic fields to constrain the particles in concentric orbits while the acceleration is accomplished by electric fields between two (or more) fixed electrodes. A typical circular resonance accelerator is the **cyclotron**. In straight-line accelerators particles are accelerated by electric fields between a large number of cylindrically shaped electrodes aligned along the axis of the beam. A simple straight-line accelerator is the **electrostatic generator**, in which an electrostatic potential difference of several million volts is subdivided between several score of tubular electrodes.

The particles which are accelerated are of two types, **electrons** and **positive ions**; the most commonly used ions are hydrogen, deuterium, or helium. Several varieties of each type of accelerator, circular or straight-line, have been developed which can accelerate either electrons or positive ions. The basic difference is that the velocity of the heavier positive ions continues to increase with increasing energy, while electrons approach the limiting velocity of light at relatively low energies ($v = 0.98c$ at 2 Mev). In circular accelerators this means that ion-rotation frequency at fixed orbit radius continues to increase with increasing energy while electrons rapidly attain constant orbital frequency. In linear resonance accelerators particle velocity determines the spacing between accelerating electrodes.

Accelerator development has proceeded in several successive stages. First came the early direct-voltage accelerators in which a d-c potential difference was developed and applied to a linear array of cylindrical electrodes within a vacuum chamber. The more successful types which have survived to the present are the voltage multiplier and the electrostatic generator.

The **voltage multiplier** uses a **Cockcroft-Walton** circuit for rectification and multiplication of voltage (see Fig. 6-1). In this circuit half-wave rectifier tubes are used with an a-c transformer to charge a capacitor; other rectifiers which become conducting in successive half cycles distribute the charge to other capacitors; ultimately the entire stack of capacitors becomes charged, with a total potential difference several times the output of the transformer, depending on the number of rectifier-capacitor units used. The original Cockcroft-Walton system provided a multiplication of 3, to a potential difference of about 0.7 million volt. This was applied to a multiple-electrode accelerating tube with the potential arranged to

accelerate hydrogen ions from an ion source in the high-voltage terminal to a target at ground potential.

With this arrangement, Cockroft and Walton at the Cavendish Laboratory in Cambridge, England, in 1932, were the first to achieve the successful **disintegration of nuclei** by electrically accelerated particles. They were awarded the Nobel Prize in Physics in 1951. Modern developments of the voltage multiplier have been

limited by voltage-insulation problems to about 1.5 million volts. Such a d-c generator can also be arranged to accelerate electrons.

The **electrostatic generator** was initiated by the experiments of **R. J. Van de Graaff** in 1930 in which he used a moving silk belt to transfer charge to a hollow metal terminal mounted on insulating supports. Continuous development since that date has led to the modern generator which is mounted in a pressure housing and uses gas at high pressure around the terminal and the belt for insulation. Figure 6-2 shows a section of a typical 4-million-volt generator of the vertical-mounting type, pioneered at the Massachusetts Institute of Technology by Van de Graaff and Trump, and now produced commercially by the High Voltage Engineering Corp. Several installations have operated at 6 million volts; the highest energy, of over 8 million volts, has been obtained at MIT. A parallel development of horizontally mounted generators

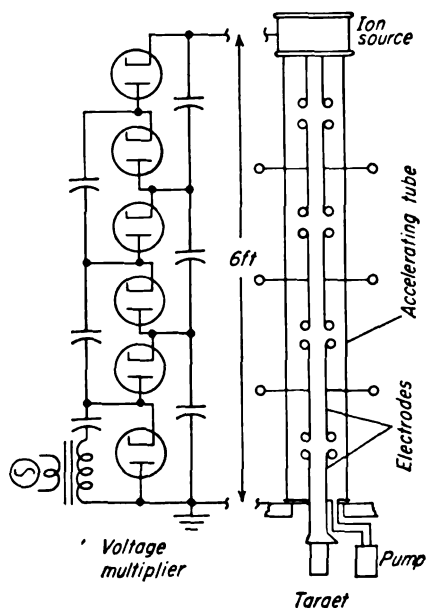


FIG. 6-1. Cockroft-Walton accelerator using a voltage-multiplier circuit producing 0.7 million volts.

tors was pioneered by Herb and his associates at the University of Wisconsin and has led to modern installations capable of operation at 4 to 6 Mev. The higher-energy machines are usually used as positive-ion accelerators for nuclear studies, while the horizontal type has found additional use as an injector or preaccelerator for still higher energy machines. Meanwhile, there has been a steady demand for low-energy (1 to 2 Mev) electron accelerators as X-ray generators for medical and industrial use.

The second stage of development, of the circular **resonance** accelerators, was instigated by the success of the **cyclotron**, developed at the University of California in 1930-1934 by Lawrence and Livingston. In resonance accelerators particles are accelerated successively in many traversals of an oscillatory electric field applied between the electrodes, attaining a final energy hundreds of times greater than the potentials applied. A simple analogy is the garden swing which can be urged to large amplitude by many successive small pushes timed to the natural period of the swing.

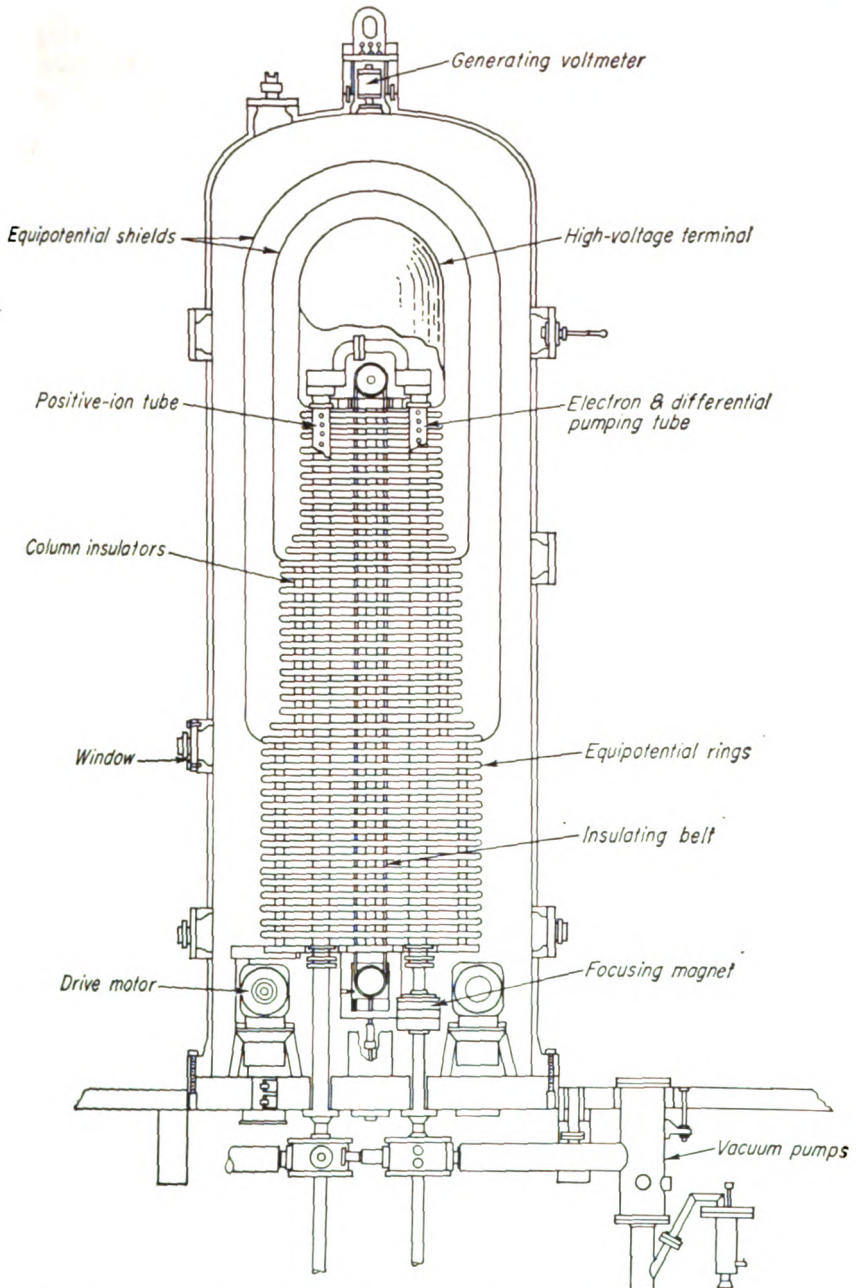


FIG. 6-2. MIT electrostatic generator for 4-million-volt positive ions or electrons.

In the **cyclotron** a large uniform magnetic field between flat cylindrical pole faces is used to provide the forces which retain the positive ions in circular orbits. A vacuum chamber fits between pole faces, and two hollow electrodes (called D's because of their shape) are mounted inside the vacuum chamber. A radiofrequency potential is applied between these electrodes so the electric field between diametral faces of the D's alternates in direction. Positive ions are produced near the center of the chamber in an ion source and are accelerated toward and into the electrode which is negatively charged at the instant. In the electric field-free region inside the D the ions are acted upon by the uniform magnetic field and travel in circular orbits inside the electrode. After traversing a semicircular path each ion returns to the diametral gap between electrodes and comes again in the influence of the electric field. For the condition of resonance, the magnetic field is adjusted so that the time required for an ion to traverse a half circle is equal to the half period of the oscillatory electric field. So on the next passage of the gap the electric field has reversed, the ion obtains another acceleration, acquires a higher velocity, and traverses a path of larger radius within the other electrode.

The force exerted on a particle of charge e and mass m , moving with velocity v in a magnetic field B which is normal to the direction of motion, is given by

$$F = evB \quad (6-1)$$

This force produces motion in a circle of radius r given by

$$\frac{mv^2}{r} = evB \quad (6-2)$$

Simplifying, we find for the angular velocity

$$\omega = \frac{v}{r} = \frac{e}{m} B \quad (6-3)$$

or, in terms of the orbital frequency f ,

$$f = \frac{\omega}{2\pi} = \frac{1}{2\pi} \frac{e}{m} B \quad (6-4)$$

This is the fundamental resonance relation for the cyclotron. As long as the mass of the particle and the magnetic field remain constant the particle will rotate at a constant frequency regardless of increasing energy and orbit radius.

The radiofrequency applied to the electrodes in a cyclotron is made equal to the ion orbital frequency. Each time an ion traverses a gap and acquires energy from the field it shifts to a path of larger orbit radius, but the frequency of rotation remains the same. The ions traverse a flat spiral path, consisting of a series of connected semicircles of increasing radius, and continue to acquire energy until they reach the outer edge of the uniform magnetic field. At this point they are deflected out of the orbit by a specially designed deflecting electric field; so they emerge from the cyclotron chamber as a beam of high-energy ions (see Fig. 6-3).

The modern fixed-frequency cyclotron has been developed to the state where it is now the work horse of nuclear laboratories. Maximum energy is about 25 Mev for deuterons or protons and 50 Mev if doubly charged helium ions (alpha particles) are used. The limitation on energy is associated with the relativistic increase in

mass of the particles with energy, such that the resonance relation [Eq. (6-4)] is not maintained. The magnet for a 20-Mev cyclotron has a 60-in.-diameter pole face and weighs about 250 tons. The largest is the Nobel Institute cyclotron at Stockholm which has an 83-in. magnet and produces 25-Mev deuterons.

Beam intensity varies from one installation to another and must be specified as to whether it represents the resonant circulating current of particles or the emergent beam deflected outside the chamber. A typical high-intensity 16- to 20-Mev cyclotron can produce a circulating deuteron beam of about 1 ma and an emergent focused beam of 0.1 to 0.2 ma. Note that this represents as much as 20 kw in the circulating beam. Such beam-power figures are significant in planning the shielding requirements for cyclotrons.

The **betatron** is an electron accelerator which uses magnetic induction for acceleration in a circular path of essentially constant orbit radius. It is not a resonance accelerator and does not use radiofrequency electric fields. A time-varying magnetic field linking the electron orbit provides an azimuthal electric field for acceleration. A weaker magnetic field at the orbit provides the central force for circular motion and also

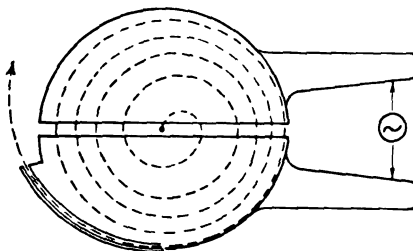


FIG. 6-3. Illustration of the ion paths within the D-shaped electrodes of a cyclotron.

increases with increasing electron energy to maintain constant orbit radius. The betatron has the advantage of resonance accelerators that high potentials are not required for acceleration, thus avoiding the limitations of insulation breakdown, and also in using a magnetic field to retain the electrons in orbits of small dimensions so the size of the accelerator is minimized. The energy gained per revolution is small and the electron makes many revolutions. However, since electron velocities are high, approaching the velocity of light at a few Mev energy, they can make many revolutions and acquire high energy in a short time. Primary credit for the development of the betatron goes to D. W. Kerst at the University of Illinois.

The time rate of change of magnetic flux linking the orbit ($d\Phi/dt$) determines the rate of increase of momentum p of the electron. In rationalized mks units, this is

$$\frac{dp}{dt} = \frac{e}{2\pi r} \frac{d\Phi}{dt} \quad (6-5)$$

For a finite flux change this relation can be integrated to give

$$p = \frac{e(\Phi_2 - \Phi_1)}{2\pi r} \quad (6-6)$$

Momentum can also be related to the field at the orbit B , from Eq. (6-3); so $mv = p = eBr$.

Inserting this relation for momentum in Eq. (6-6), we obtain

$$(\Phi_2 - \Phi_1) = 2(\pi r^2)B \quad (6-7)$$

This relation shows that the flux change linking the orbit must be twice the value

obtaining if flux density were uniform and equal to the field at the orbit. It is the well-known "2:1 rule" for acceleration at constant orbit radius in the betatron.

Simple betatrons are built with laminated iron as in a transformer, and operate on 60- or 180-cycle alternating current. Figure 6-4 illustrates such a simple machine, showing a doughnut-shaped vacuum chamber between pole pieces at the orbit and a central core of iron to provide the linkage flux. A large number of such simple machines have been built and sold for industrial and medical uses by industrial firms, for electron energies of about 20 Mev. The simplicity of the machine is illustrated by the use of sealed-off glass or ceramic vacuum chambers, in which a hot cathode provides the electrons for acceleration.

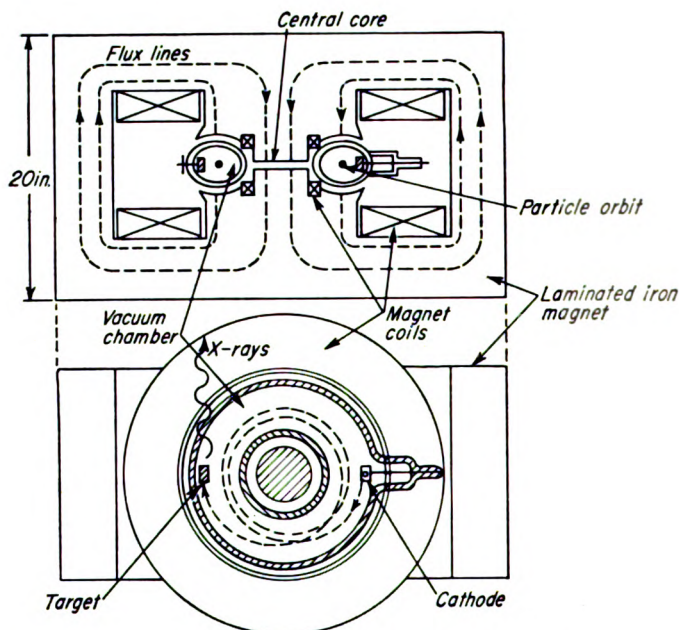


FIG. 6-4. Magnetic-circuit arrangement for a simple betatron, showing the doughnut-shaped vacuum chamber.

High-energy betatrons operate on pulsed power and require complicated power supplies and switching circuits. They have also been subject to a long and detailed program of model studies, magnetic field corrections, and development of control circuits. The largest is the 300-Mev betatron built by Kerst at the University of Illinois. With the development of the synchrotron, described below, it has become possible to attain similar or higher electron energies with a much simpler and cheaper ring magnet; so few high-energy betatrons have been built.

The third phase of accelerator development has utilized the principle of **phase stability in synchronous acceleration**. This principle was announced simultaneously and independently by McMillan at the University of California and by Veksler of Moscow, U.S.S.R., in 1945. It offers a mechanism by which the particles in reso-

nance accelerators can be kept in phase with the accelerating radiofrequency voltage for many thousands of small accelerations.

When a particle moves in resonance with the oscillations of an electric field the phase of the field at the instant the particle crosses the accelerating gap determines the energy transmitted to the particle. In circular magnetic accelerators the orbits are circular and the accelerating gaps are at fixed locations in the orbit. In Fig. 6-5 the voltage across such an accelerating gap is plotted as a function of time. A resonant particle crossing at zero phase is not accelerated and arrives at the next gap also at zero phase. A particle crossing at time t_1 , at a phase angle ϕ_1 , acquires energy, traverses a circle of larger radius, and requires a longer time to reach the next gap. This shift in phase will continue until the particle phase shifts into the region of deceleration, where the effect reverses and the particle is reduced in energy. This results in stable oscillations in phase of the particle about the zero-phase position. Such a particle could, in principle, remain indefinitely oscillating about this fixed equilibrium energy and the associated equilibrium orbit.

Several mechanisms are available for disturbing the equilibrium conditions described above so the particle can be continuously accelerated. Each mechanism has led to a new type of accelerator, of which the **synchrotron**, the **synchrocyclotron**,

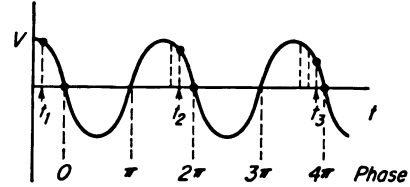


FIG. 6-5. Phase diagram for synchronous acceleration. Particles crossing the accelerating gap at random phase migrate toward the equilibrium-phase position.

the **proton synchrotron**, and the **synchronous linear accelerator** are the most important examples.

The **electron synchrotron** uses a magnetic field which increases with time and which retains the electrons in a circular path of essentially constant radius. Electrons are accelerated by a fixed-frequency oscillating electric field applied across a gap in the conducting lining of the vacuum chamber at one or more points in the particle orbit. The rising magnetic field produces a tendency for the electron orbit to contract and for the frequency of rotation to increase, thereby causing a shift in phase of crossing the gap. This shift in phase

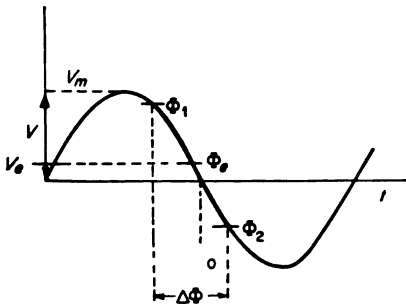


FIG. 6-6. Superposed phase diagram illustrating phase oscillations about an equilibrium phase, and showing the average value of volts per turn available for acceleration.

means that the electrons acquire more energy, sufficient to compensate for the rising magnetic field. The phase oscillations described above occur about an equilibrium phase and an associated average acceleration rate which is determined by the rate of rise of magnetic field. Figure 6-6 illustrates this oscillatory motion by superposing the phase diagrams covering a complete phase-oscillation cycle. The particle oscillates between limits illustrated by the arbitrary phases ϕ_1 and ϕ_2 in the figure,

requiring some thousands of revolutions to complete a phase cycle, and gaining an average increment of energy in each turn given by the equilibrium value of volts per turn. The volts per turn V_e is related to the rate of rise of magnetic field as

$$V_e = 2\pi r^2 \frac{dB}{dt} \quad (6-8)$$

The similarity of this behavior to that of a synchronous motor suggested the name of the machine.

Maximum energy depends on orbit radius and on peak magnetic field. The relation in mks units is

$$T = ceBr = 300Br \quad (6-9)$$

To find the energy in Mev units, use the constant on the right with B in webers/m² and r in meters. The orbital frequency of the electrons is given by

$$f = \frac{c}{2\pi r} = \frac{47.8 \times 10^6}{r} \quad \text{cycles/sec} \quad (6-10)$$

The radiofrequency accelerating field is made equal to this orbital frequency and is applied across the gap in the orbit by using a quarter-wave resonant cavity circuit.

Electrons do not reach the velocity of light and so do not have the equilibrium orbit radius until they have acquired several Mev energy. So a preliminary "betatron start" phase of operation is generally used to accelerate the electrons to about 4 Mev.

The first group of synchrotrons to be brought into operation were designed for the energy range 300 to 350 Mev. The primary limitation was the heat loss in the iron and copper at the required high flux densities under continuous a-c excitation. Accordingly, most of them were designed to be pulse-powered for only a few pulses per second. A typical pulse-power system uses a capacitor bank to store the energy and ignitron switching tubes to connect the capacitor to the magnet and to disconnect after one oscillatory cycle. A typical synchrotron is illustrated in Fig. 6-7, showing the MIT 330-Mev machine which has an orbit radius of 1 m. It operates at 6 pulses/sec, accelerating about 1×10^8 electrons/pulse.

The **synchrocyclotron** uses a variation in frequency of the accelerating electric field to maintain synchronous acceleration. It is basically a cyclotron, with a uniform magnetic field over a large area between cylindrical poles, and a D-shaped electrode which develops the radiofrequency accelerating field across a diametral gap. With constant-frequency acceleration the protons would show an appreciable relativistic increase in mass at about 20 Mev energy and the rotation frequency would decrease following the resonance relation of Eq. (6-4). By continuously and steadily decreasing the applied frequency the particles will remain in the accelerating phase and continue to be accelerated. The time rate of change of frequency determines the rate at which energy must be supplied to maintain resonance, and so determines the necessary volts per turn. If the peak radiofrequency voltage available is adequately larger than this equilibrium value of volts per turn, the particles will perform phase oscillations about the equilibrium phase, gaining first more and then less than the equilibrium value. Acceleration will continue to an orbit radius corresponding to the physical limit of the uniform magnetic field.

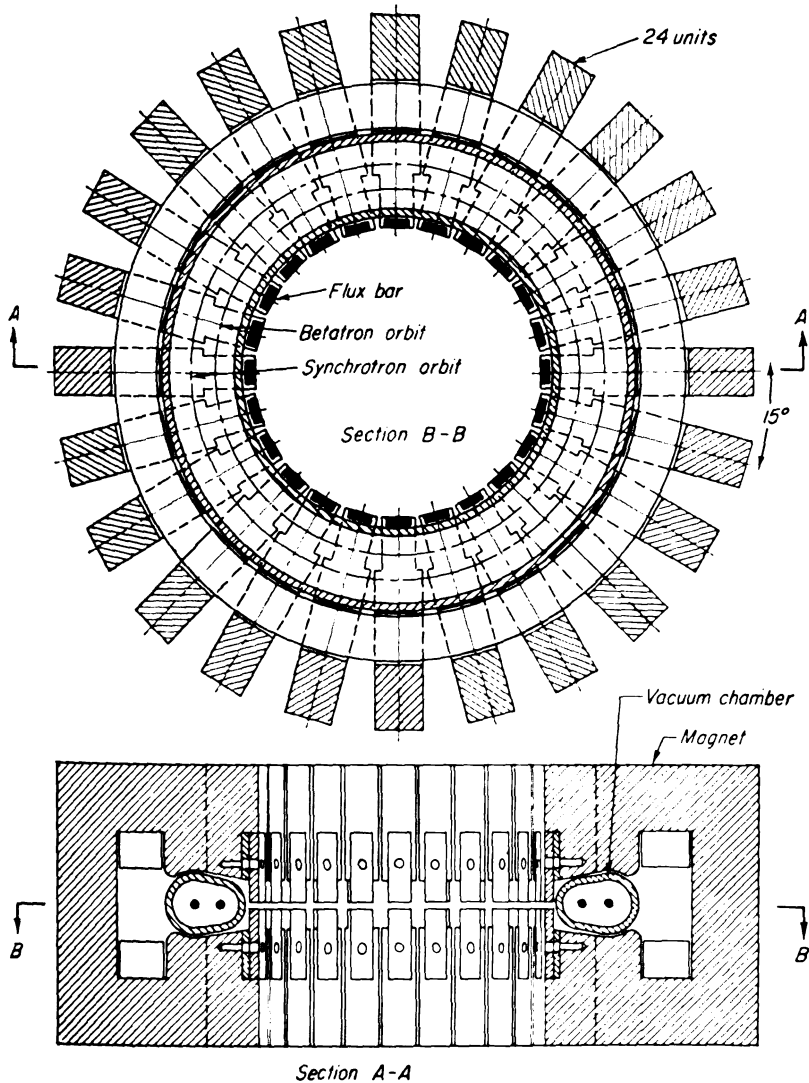


FIG. 6-7. MIT 330-Mev electron synchrotron.

Frequency of the resonant D circuit is varied by a rotating capacitor in the circuit, which produces a cyclic change in frequency. In each decreasing frequency sweep a bunch of resonant ions is accelerated to larger radii and higher energy; so the output is a sequence of pulses of high-energy protons at the frequency of the modulating system. The larger the modulation frequency the higher is the number of pulses per second and the average beam intensity. Figure 6-8 is a schematic plan view of a synchrocyclotron, illustrating one method used to obtain an emergent ion beam.

About eight medium or large synchrocyclotrons have been built in this country and a similar number abroad. Those designed for less than 200 Mev are used for extending to higher energy the results of nuclear researches of the type done with standard cyclotrons. The higher-energy machines are of primary value as sources of mesons for high-energy particle research. The highest-energy machine now in operation is the 680-Mev accelerator at the U.S.S.R. Academy of Sciences laboratory near Moscow.

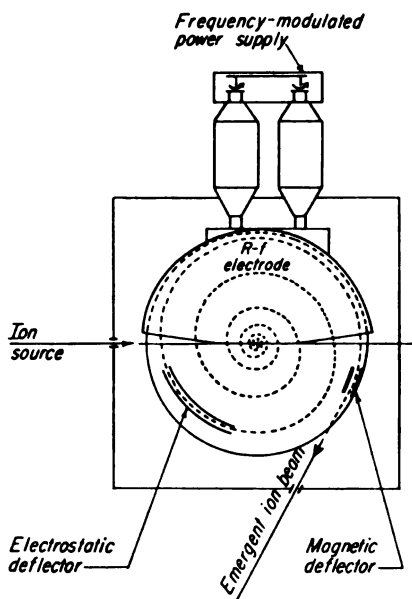


FIG. 6-8. Schematic view of a synchrocyclotron chamber showing the frequency-modulated r-f circuit and an arrangement for obtaining an emergent beam.

One of the most productive synchrocyclotrons in this country is the 450-Mev machine at the University of Chicago. It has a pole-face diameter of 170 in. and a magnet weighing 2,200 tons. The time average beam current is $1 \mu\text{a}$; so the maximum accelerated beam power is 0.5 kw.

The **proton synchrotron** is the largest and highest-energy type of accelerator. The **cosmotron** at Brookhaven National Laboratory on Long Island and the **bevatron** at the University of California in Berkeley are the two largest examples. The cosmotron produces 3 Bev (3,000 Mev) protons and the bevatron develops 6 Bev. These energies exceed the rest mass of a proton or a neutron, which is about 0.94 Bev, and of all the known mesons and other strange particles observed in cosmic-ray studies. Processes can be studied in the laboratory in which nucleons are created from energy, such as the negative proton discovered in Berkeley in 1955, and the antineutron in 1956.

The principle of operation is basically the same as for the electron synchrotron. A ring-shaped magnet produces a magnetic field in a vacuum chamber between the pole faces, and the magnet is pulsed so the field increases with time as the protons gain energy, so as to maintain constant orbit radius. However, the velocity and the frequency of rotation of the protons increase during the acceleration; so the applied radiofrequency accelerating field must be modulated to match the increasing orbital frequency in order to maintain synchronism. The required schedule of frequency modulation does not follow any simple law but depends on the rate of increase of magnetic field. Protons are first accelerated to several Mev energy in an auxiliary accelerator (4-Mev Van de Graaff generator for the cosmotron; 10-Mev proton linac for the bevatron) and then inflected into the orbit at the instant the magnetic field reaches the correct value for this energy. A plan view of the cosmotron is shown in Fig. 6-9.

The rate of pulsing of these large extended magnets is slow, once every 5 sec for the cosmotron, and the time for accelerating to high energy relatively long (1 sec).

In each pulse about 2×10^{10} protons are brought to high energy. The average beam current has little significance. The average power in the beam at 3 Bev energy is about 2 watts, which is one thousandth of that from a large synchro-cyclotron.

Within the general category of straight-line accelerators the name **linear accelerator** has been applied to one specific type, in which the acceleration is accomplished by radiofrequency electric fields, of which there are two modifications applying to proton accelerators (the **linac**) and to electron accelerators. They are fundamentally different from the circular magnetic accelerators. However, they also use the principle of phase-stable or synchronous acceleration, at least in the

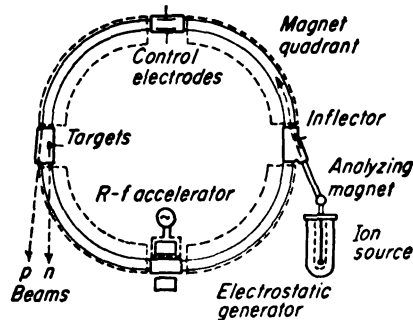


Fig. 6-9. Plan view of cosmotron magnet and assembly.

early stages of acceleration. The chief advantage of linear accelerators lies in the excellent collimation of the emergent beams, which have small diameters and high densities. They have the disadvantages of a rather broad spread in energy of the beam and higher cost due to the extensive power sources required. However, if sufficient power is provided, beam intensities can be obtained which are higher than in circular machines.

The first successful proton **linac** was completed by Alvarez at the University of California in 1948, producing a beam of 32-Mev protons with an average current of $0.4 \mu\text{a}$. Difficulties were met and solved in focusing the beam and in developing the 200-megacycle power supply. Several other linacs are now under construction based on the Berkeley experience, with energies of 50 to 80 Mev. Some of these machines are to be used as injectors for the superenergy alternating-gradient accelerators of the future (see Ultra-high-energy Accelerators).

The **electron linear accelerator** has reached its highest stage of development at Stanford, as the culmination of a long series of developments of klystron power tubes, microwave waveguides, and other components by the late W. W. Hansen, assisted by Ginston, Panofsky, and others. The Stanford machine is a 220-ft waveguide at 10.7 cm wavelength driven by 15 high-powered klystrons. It has produced protons of 600 Mev energy, with an average current of $1 \mu\text{a}$ and an average power in the beam of 0.6 kw. Figure 6-10 shows a typical section of 10.7-cm waveguide for such a linear electron accelerator. Figure 6-11 is a schematic illustration of the fields in the waveguide and also illustrates the phase focusing or bunching of the electrons during acceleration.

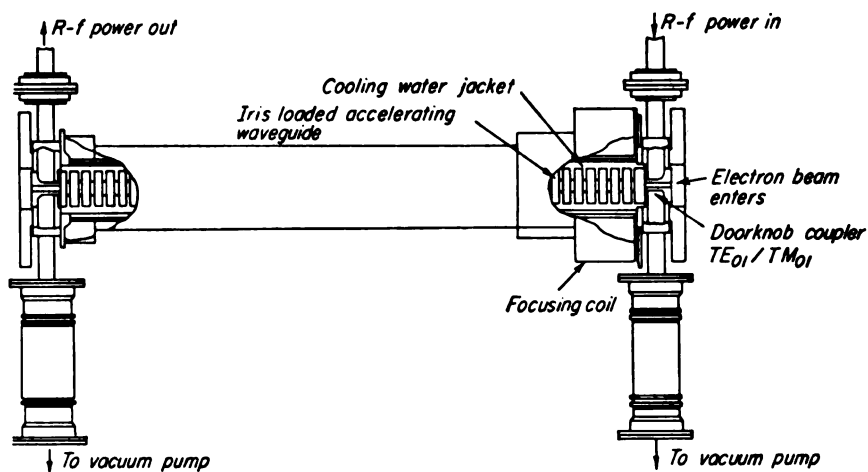


FIG. 6-10. Section of microwave waveguide for an electron linear accelerator. (Courtesy of High Voltage Eng. Corp.)

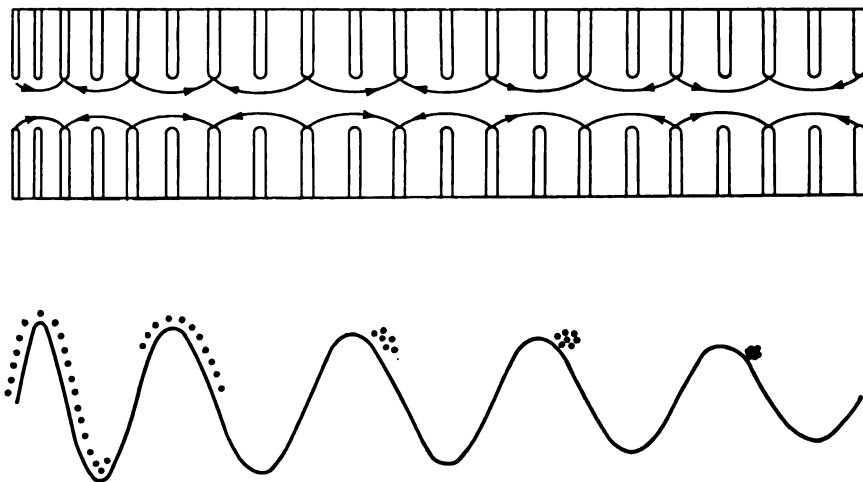


Fig. 6-11. Electric field inside an electron linear accelerator waveguide, and an exaggerated illustration of phase bunching.

RADIATIONS FROM ACCELERATORS

Many different radiations are emitted from accelerators, differing in kind and in relative intensity in the several machines. Some of these radiations are of such short range or of such low intensity that they are of interest only to nuclear scientists. We are interested here primarily in those radiations which are produced in sufficient intensity and are sufficiently penetrating to be significant in the problem of shielding. The sources and chief characteristics of these radiations are described below.

Primary radiations such as protons, deuterons, or electrons are frequently brought outside the accelerator vacuum chamber through thin-foil windows or in evacuated

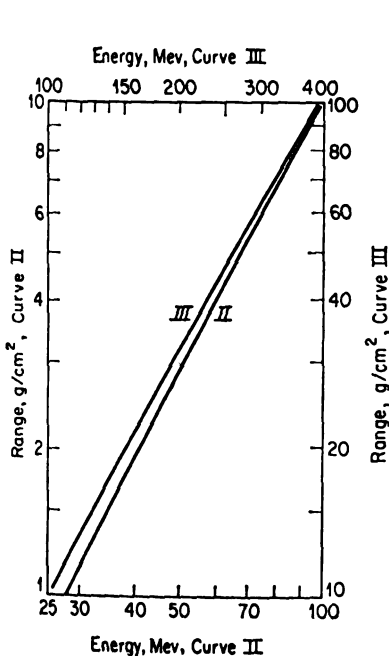


FIG. 6-12. Mean range of protons in aluminum (in g/cm²) for proton energies between 25 and 400 Mev. (Brookhaven Natl. Lab. Rept. BNL-T-7.)

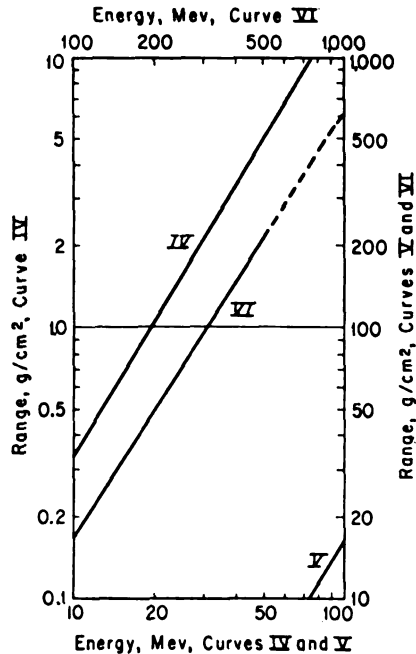


FIG. 6-13. Mean range of protons in lead (in g/cm²) for proton energies between 10 and 1,000 Mev. (Brookhaven Natl. Lab. Rept. BNL-T-7.)

pipes. Some of these radiations are quite penetrating, especially for the higher-energy accelerators. However, for most machines the primary radiation can be absorbed in a relatively thin metal chamber wall. For example, 20-Mev protons have a range of 4 m in air or of 1 mm in lead. Laboratory practice requires careful planning to avoid exposure of personnel to either the primary beam or scattered radiation. Several cases exist where scientists have suffered burns of the hands or arms from scattered beam radiation. In general, however, this is a problem in laboratory arrangements rather than a shielding problem.

The high-energy protons from synchrocyclotrons are more penetrating. Range-energy curves for protons in air, aluminum, copper, silver, and lead have been computed by Bethe from basic ionization-loss theory, and were published in a Brookhaven Report* in 1949. These ionization-loss curves are still valid for proton energies below a few hundred Mev, but at higher energies nucleonic interactions such as meson production must be considered and range is no longer a satisfactory measure of penetration. Figure 6-12 taken from Bethe's curves gives the range in aluminum (in g/cm²) of protons from 25 to 400 Mev energy. Figure 6-13 gives the range of protons in lead from 10 to 500 Mev. In estimating absorber thickness it is also necessary to consider the range straggling about the mean ranges, also given in the Brookhaven Report. Straggling extends the range about 20 per cent at low energies, dropping to about 10 per cent at the highest energies.

Range is not a good measure of electron penetration. Absorption is primarily through electromagnetic interactions such as the production of X-rays, which in turn produce secondary electrons and positron-electron pairs. These phenomena are described under a following section on X-rays.

Gamma rays are produced by excitation of nuclear-energy levels. The line spectra observed are characteristic of the particular nucleus involved, and the relative intensity of the lines varies with the degree of excitation. The target nucleus and the nuclear interaction involved determine the spectrum. Since many targets will be used in an accelerator it is useful in planning the shielding to consider a composite spectrum covering a large number of targets, incident particles, and reactions. Such a composite spectrum is illustrated in Fig. 6-14, developed from the following considerations:

(1) the average spacing between low-lying energy levels in most nuclei is less than 0.1 Mev; (2) a majority of the targets used for nuclear research are of low atomic number, for which the excitation-level spacing is about 0.5 Mev; (3) gamma rays occur primarily in cascade steps between energy levels, with low probability for high-energy quantum emission; (4) most nuclei become unstable against neutron emission for excitations greater than 7 Mev.

The attenuation of such a composite spectrum in absorbers does not follow any simple absorption law, but exact calculations can be made for any relatively narrow energy band and the results can be summed to provide an estimate of the over-all attenuation. For most shielding problems a simple approximation is adequate, which is to assume that the dominant energy band is between 2 and 4 Mev energy, for which the mass-absorption coefficient has its minimum value of 0.45 g./cm² in Pb. Shielding requirements based on the estimated intensity of gamma rays in this 2-

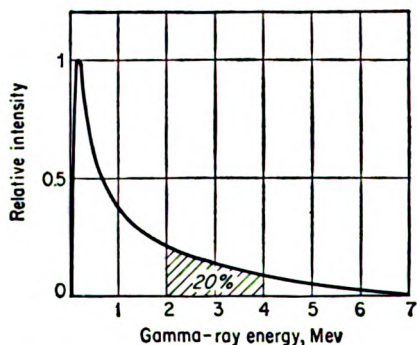


FIG. 6-14. Composite spectrum of nuclear gamma rays. The most penetrating component (in Pb) is the energy band between 2 and 4 Mev.

* Bethe, Range-energy Curves, *Brookhaven Natl. Lab. Rept.* BNL-T-7, June 1, 1949.

to 4-Mev band will in general prove adequate for the remainder of the spectrum. The fraction of gamma-ray intensity within this band is about 0.2 for the composite spectrum shown in Fig. 6-14.

Cyclotron gamma rays emerge from the target in an almost isotropic distribution but are heavily shadowed by the target chamber and the magnet poles; so the highest intensity is in the plane of the magnet gap. Gugelot and White,* in their study of neutron shielding, used the gamma rays from 16-Mev protons on Be and measured the attenuation in water, concrete, and several types of high-density concrete. They observed absorption coefficients which decreased with absorber thickness, indicating a progressive hardening of the radiation compatible with the energy distribution described above. The terminal half-value-layer is reported to be 13.3 cm in ordinary concrete (density 2.35); this corresponds to an **absorption coefficient of 0.052 per centimeter**, which represents the value for 12-Mev monoenergetic X-rays in concrete (see Fig. 6-15). Delano and Goodman† used the gamma rays from 15-Mev deuterons on Be and observed a terminal **absorption coefficient of 0.068/cm** in concrete of density 2.20, characteristic of 10-Mev X-rays. An explanation of these abnormally low absorption coefficients is that other gamma rays are generated in the shield by neutrons. In fact, Delano and Goodman distinguish three sources of gamma radiation: (1) primary gamma rays from the target, (2) gamma rays from thermal-neutron capture by hydrogen and other materials in the shield, and (3) secondary gamma rays from radioactive products of neutron reactions in the shield. The neutron-produced gamma rays are formed throughout the shield and extend the gamma-ray intensity abnormally. If we assume an effective mean energy of 5 Mev for the distribution given in Fig. 6-14, we find (Fig. 6-15) an **absorption coefficient of 0.068 per centimeter in concrete**. In the absence of neutrons we would expect the gamma rays to be attenuated with this coefficient. However, the cyclotron radiation is a mixture of neutrons and gamma rays. Gamma-ray intensity is observed to build up in the first 20 cm of concrete, paralleling the build-up of thermal-neutron intensity. Neutron-produced gamma rays represent a significant component. The apparent terminal absorption coefficient represents the observed attenuation of the total gamma rays present. The absorption coefficient measured by Gugelot and White of 0.052 per centimeter is equiv-

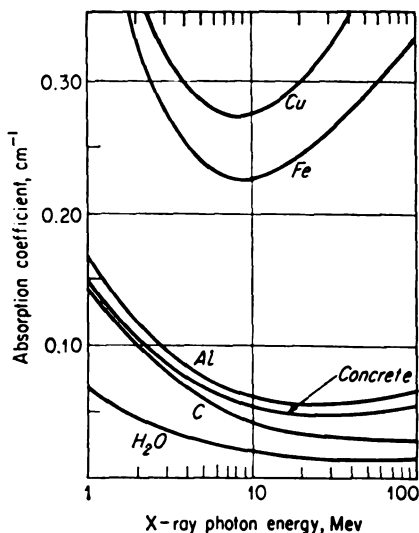


FIG. 6-15. Absorption coefficients for monoenergetic X-rays in several materials as a function of X-ray energy between 1 and 100 Mev. (From Fig. 9, *NBS Handbook 55*, Feb. 26, 1954.)

alent. The neutron-produced gamma rays are formed throughout the shield and extend the gamma-ray intensity abnormally. If we assume an effective mean energy of 5 Mev for the distribution given in Fig. 6-14, we find (Fig. 6-15) an **absorption coefficient of 0.068 per centimeter in concrete**. In the absence of neutrons we would expect the gamma rays to be attenuated with this coefficient.

However, the cyclotron radiation is a mixture of neutrons and gamma rays. Gamma-ray intensity is observed to build up in the first 20 cm of concrete, paralleling the build-up of thermal-neutron intensity. Neutron-produced gamma rays represent a significant component. The apparent terminal absorption coefficient represents the observed attenuation of the total gamma rays present. The absorption coefficient measured by Gugelot and White of 0.052 per centimeter is equiv-

* *J. Appl. Phys.*, vol. 21, p. 369, 1950.

† *J. Appl. Phys.*, vol. 21, p. 1040, 1950.

alent to an attenuation factor of 4.8 per foot of concrete of density 2.35. This is considerably smaller than the measured attenuation of fast neutrons in concrete (see below), from which the conclusion is often drawn that the gamma radiation is more penetrating than the neutron radiation from cyclotrons. It should be noted that this is only an apparent result, due to neutron-produced gamma rays in the absorber; so the problem of gamma-ray shielding cannot be separated from neutron shielding.

The use of high-density iron-loaded concrete is known to increase absorption coefficients for both neutrons and gamma rays. Boron is also used to capture slow neutrons without emission of gamma rays. Gugelot and White* measured the absorption in several types of loaded concrete. It is noteworthy that the strongest neutron absorption was observed with limonite-ore concrete (density 2.63), which has a high concentration of water, while the strongest gamma-ray absorption came from a more dense (3.60) concrete including iron and pyrex (boron). In all cases, however, the half-intensity thickness for gamma rays exceeded that for neutrons. The relative incident intensities of neutrons and gamma rays will determine which type of shield is most efficient. Furthermore, the relative biological effectiveness (RBE) of gamma rays is smaller than for neutrons; so the emergent gamma-ray intensity level can be higher. The conclusion is that either ordinary concrete or high-density concrete can be used for cyclotron shielding, with the choice determined by total radiation intensity and space considerations.

X-rays are the major radiation from electron accelerators, emitted from all targets bombarded by the electron beam. They are also a problem in d-c positive-ion accelerators, coming from backstreaming secondary electrons impinging on the ion source. Lead shielding of the ion source is a necessity in many electrostatic-generator installations, for example.

The absorption coefficients for monoenergetic X-rays can be computed from the classical theory of electromagnetic interactions, and are confirmed by many experimental measurements. At low energies (<3 Mev) photoelectric absorption and Compton scattering are the significant interactions; at higher energies **electron-positron pair formation** becomes dominant. This leads to a minimum in the curve of absorption coefficient vs. energy, with the location of the minimum a function of atomic number of the absorber; the minimum occurs at 3 Mev for Pb, 8 Mev for Cu, and 20 Mev for Al. Figure 6-15 is a plot of the monoenergetic X-ray absorption coefficients for several materials as a function of energy, taken from NBS Handbook 55 (see Sec. 3).

The X-rays emitted from targets bombarded by high-energy electrons, called **bremsstrahlung**, have a continuous distribution in energy extending up to the maximum energy of the electrons. The spectrum can be computed from the theory of electromagnetic interactions originally developed by Bethe and Heitler, and more recently by Schiff.† Schiff's results have been computed and tabulated in the NBS Handbook, giving the relative number of photons in the forward direction for a sequence of incident electron energies up to 100 Mev. Figure 6-16 is a plot of the X-ray spectrum in the forward direction for 50- and 100-Mev electrons.

The **angular distribution** of the X-rays from an accelerator target is primarily due

* *J. Appl. Phys.*, vol. 21, p. 369, 1950.

† *Phys. Rev.*, vol. 83, p. 252, 1951.

to the multiple scattering of the electron beam in the target. For a thin target where multiple scattering is negligible the forward momentum of the electrons produces an angular distribution in which about half the intensity is included within a cone of half angle given by mc^2/E , the ratio of rest energy to total energy of the electron. At 300 Mev, for example, half the intensity is contained in a cone of half angle 0.4° . Multiple scattering in a thick target broadens the distribution to several times this angle.

The attenuation of the continuous spectrum of bremsstrahlung has been measured for several energies with the 50-Mev betatron at the National Bureau of Standards, using ordinary concrete as the absorber. The result is surprising, in that the log-intensity plots are essentially straight lines for all energies between 6 and 38 Mev. The absorption coefficient observed at 30 and 38 Mev is about 0.14 per inch of concrete, shown to be valid for electron energies

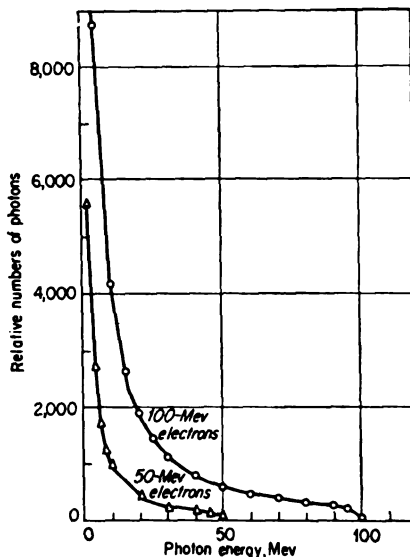


FIG. 6-16. Continuous X-ray spectrum (bremsstrahlung) from electrons of 50 and 100 Mev energy.

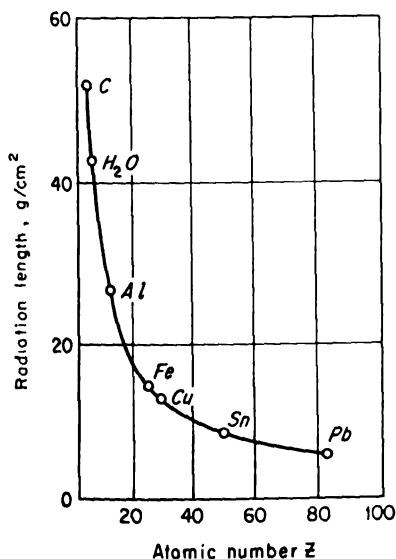


FIG. 6-17. Radiation lengths in g/cm^2 as a function of atomic number.

between 30 and 100 Mev. This must not be confused with the linear absorption observed with monoenergetic radiation. It is an "effective" coefficient which is a consequence of the exponential shape of the bremsstrahlung spectrum (Fig. 6-16).

A useful quantity in X-ray attenuation theory is the **radiation length**, $X_0 = 1/N\varphi_{\text{rad}}$, where N is the number of atoms per cubic centimeter and φ_{rad} is the probability of energy loss by radiation by an electron, per centimeter of path. It is the mean free path for energy loss ($E = E_0/e$) by an electron through radiation. It is useful in that the fractional energy lost by an electron per radiation length is almost independent of target material and electron energy. Figure 6-17 is a plot of radiation lengths in g/cm^2 for a number of elements as a function of atomic number, taken from NBS Handbook 55 (see Sec. 3).

The process of absorption of radiation of 100 Mev (or higher) is known as a "soft shower." The dominant X-ray interaction is pair production; secondary electrons and positrons produce other lower-energy X-rays (and also positron-annihilation gamma rays of 0.5 Mev). The multiplication process builds up the total number of photons and electrons in the shower, and ionization intensity increases in the first layers of the absorber. Then, after a "pseudo equilibrium" is established for the continuous spectrum, the total intensity attenuates with the same "effective" absorption coefficient which applies in the 30- to 100-Mev region. The value given above for concrete (0.14 per inch) is 0.13 g/cm² of concrete, or when converted to radiation length units is **0.0047 radiation length**. The same value is valid for other materials, if measured in radiation lengths, and can be converted to g/cm² units by using Fig. 6-17.

For very high energy showers and great thicknesses of absorber the "effective" absorption coefficient no longer applies. A new and final type of equilibrium becomes established, in which most of the X-rays are reduced to the true minimum ionization energy illustrated in Fig. 6-15. This minimum absorption coefficient occurs at different energies in different absorbers. In a high-energy electron-photon shower about 20 radiation lengths is required for such an equilibrium to be established. Beyond this, attenuation is governed by the true minimum absorption coefficient, which is roughly a factor of 10 per 6 radiation lengths, or an **absorption coefficient** in radiation length units of 0.38 per radiation length.

High-energy X-rays and gamma rays also produce neutrons through the photo-production processes, (γ, n) , $(\gamma, 2n)$, etc. The cross section is small, about one hundredth that for neutron production by protons (see following discussion), and neutron dose rate in the forward direction is insignificant in comparison with the X-ray dose rate. In the forward direction the shielding required for X-rays is more than sufficient to absorb the photoneutrons. However, the photoneutrons have a very broad angular distribution which is very nearly isotropic; so the neutron dose rate at wide angles is comparable with the scattered X-ray dose rate. If the electron energy is high and the intensity is also large, the photoneutrons may become the dominant radiation in the direction of the sides and roof of the accelerator building. With the Stanford 600-Mev linear accelerator* the intensity of photoneutrons above and on the sides of the target areas is sufficient to require about 5 ft of concrete shielding.

Neutrons are produced from practically all types of accelerators and in a wide variety of nuclear reactions. Yields depend strongly on the incident particle energy and on the particular reaction. Neutron energy also varies widely, and there is a characteristic energy distribution from each reaction. Since many targets will be used in a laboratory program, and different incident particles, it is wise to base the shielding designs on the highest-yield process envisaged. It is customary to discuss the shielding problems in terms of several distinct energy bands, partly because they represent the output of different accelerator types and partly because of their different characteristics in absorption. These energy bands can be listed:

* Chodorow, Ginzton, Hansen, Kyhl, Neal, and Panofsky, *Rev. Sci. Instr.*, vol. 26, pp. 134-204, 1955.

1. Thermal and resonance neutrons, 0 to 1 ev
2. Low-energy "fast" neutrons, 0.1 to 10 Mev, from
 - a. Nuclear interactions of positive ions
 - b. "Giant-resonance" neutrons from photonuclear processes
3. High-energy neutrons, about 100 Mev or higher

The properties of pile-produced neutrons are described under Nuclear Reactors in this section, and in Secs. 7 and 8, including the shielding requirements. Accelerator-produced neutrons have much the same properties, except for the initial energy distribution. The same processes apply, such as elastic and inelastic scattering in an absorber, slowing down to thermal energies, diffusion of thermal neutrons, and the capture of thermal neutrons with emission of radiation or formation of radioactive products. The properties of **thermal and resonance neutrons** are identical with those from a pile and will not be repeated here.

Positive-ion accelerators up to 20 Mev energy produce **fast neutrons** with an energy distribution extending to considerably higher energies than pile-produced neutrons. In this energy range the most probable neutron-emitting process is that of deuteron stripping, $X(d,n)$, and the highest yields are observed with the lightest targets such as D-2, Li-6, 7, and Be-9. Neutrons have been observed from the D-2(d,n) reaction at incident deuteron energies as low as 30,000 ev. Yield increases with energy up to the nuclear potential barrier for deuterons and continues to increase at higher energies because of increased penetration into the target if thick targets are used. Yield is also directly proportional to beam intensity. A practical intensity limit is frequently set by melting or evaporation of the target (heavy-water ice has been used, for example). The most rugged target developed to date is water-cooled beryllium, and maximum neutron intensities are observed from the Be(d,n) reaction with cyclotrons and with the target in the high-intensity internal beam. With 15-Mev deuterons the **thick-target** yield is about 1 neutron per 200 deuterons. At this energy a 1-ma internal beam represents 15 kw heating in the target. If this could be cooled it would produce 3×10^{13} neutrons/sec. However, few cyclotrons have been equipped with targets capable of such high-intensity operation.

It might be noted that, although the neutron yield from a cyclotron is small compared with a high-flux pile, the **fast-neutron flux** (neutrons/cm²/sec) behind the target exceeds the flux in a pile by several orders of magnitude, because of the small dimensions of the beam at the target.

The **neutron energy distribution** from 16-Mev protons on a thick Be target was measured by Gugelot and White* and is reproduced as Fig. 6-18. This distribution has a peak at 2 to 3 Mev and a tail extending up to 9 Mev. Other reactions will produce different energy spectra. Delano and Goodman,† using 15-Mev deuterons on Be, studied the attenuation in concrete of fast neutrons of 4.5, 7.6, and 8.3 Mev energy, by using detectors depending on the activation of aluminum and iron foils through the reactions Al(n,p), Fe(n,p), and Al(n,α), but did not measure the neutron energy distribution. In several cyclotron laboratories shielding requirements have been based on an assumption that the average initial energy was 4 or 5

* *J. Appl. Phys.*, vol. 21, p. 369, 1950.

† *J. Appl. Phys.*, vol. 21, p. 1040, 1950.

Mev. The higher initial energy causes somewhat greater penetration than for pile-produced fast neutrons. However, the additional penetration is not greater than an absorber thickness equivalent to one mean free path for the fast neutrons in the absorber.

The experimental results of Delano and Goodman on the **fast-neutron** components mentioned above lead to an attenuation factor of 8.3 per foot of concrete. Similar measurements of thermal and resonance neutrons using indium foils give an attenuation of 10 per foot. They conclude that a reasonable value for shielding-

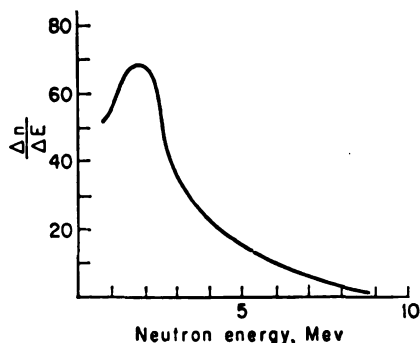


FIG. 6-18. Neutron energy distribution from a cyclotron with 16-Mev protons on a Be target.

design purposes is an attenuation factor of 9 per foot of ordinary concrete, for cyclotron neutrons. This is equivalent to a half-value layer of 2.7 in. The detectors used in these measurements were specific for neutrons, and the results do not include the transformation of neutron energy into gamma radiation discussed elsewhere. Again, the two radiations should not be considered separately. Neutron-produced gamma rays in the absorber extend the observed penetration of the gamma rays. The half-value layer given above is in fact smaller than would be observed in the absence of gamma-ray production. As a consequence it seems necessary, for a

"pure" neutron source, to use a larger half-value layer of the order of 3.0 in. of concrete which is comparable with the value accepted for pile-produced neutrons. This is equivalent to an absorption coefficient of 0.23 per inch in concrete.

Gugelot and White further studied the attenuation of neutron intensity in several types of high-density concrete. The half-value layer for neutrons was smallest for a limonite-ore concrete with a high water content. However, gamma-ray attenuation is also important, and for gamma rays maximum iron content and maximum density are preferable. Again, the choice of materials will be primarily determined by space considerations.

Electron accelerators produce fast neutrons through the **photonuclear processes** (γ, n), ($\gamma, 2n$), (γ, np), etc., in which an X-ray is the incident radiation. The simple photoneutron process (γ, n) has been observed for a wide range of target elements, with an X-ray threshold energy varying between 8 and 16 Mev. Most of the studies have been made with betatrons and electron linear accelerators of about 20 Mev energy. At these energies the yields are small, varying between 10^5 and 10^7 neutrons per mole of target material for 1 r of primary X-ray beam exposure (see NBS Handbook 55). The shielding provided for the X-radiation is more than adequate for the neutrons.

Photons excite nuclei to excitation states quite similar to those produced in positive ion reactions, and the neutron energy spectrum is also similar to the cyclotron neutron spectrum discussed elsewhere. The cross section for photoproduction of neutrons shows a broad **giant resonance** for photon energies of about 20 Mev,

associated with a maximum in the density of nuclear levels in nuclei. Yield increases with electron energy because of the increased intensity of the 20-Mev band in the bremsstrahlung spectrum (see Fig. 6-16). The yield has been estimated by Chodorow *et al.* and can be expressed most simply as $0.4E_0$ neutrons per incident electron, where E_0 is electron energy in Bev units. Thus, at the Stanford linear accelerator which operates at 600 Mev with an average current of $1 \mu\text{a}$, the neutron yield is about 10^{12} per second.

The significance of **photoneutrons** in shielding is that they have an almost isotropic distribution. In the forward beam direction the shielding for X-rays is always adequate for the neutrons. However, for wide-angle shielding the neutron component is dominant. For shielding calculations the Stanford group assume an effective average neutron energy of 9 Mev, for which they take a half-value layer thickness in concrete of 3.0 in.

High-energy electron accelerators produce, in addition to the "giant-resonance" neutrons, another broad group of **"high-energy" neutrons** of about 100 Mev energy. These are presumably the high-energy tail of the direct photoeffect neutrons (γ, n), attributed by Levinger * to "quasi deuterons" in the nucleus undergoing photodisintegration. This suggests that the yield of neutrons should be similar to that of high-energy protons, for which some measurements are available. Estimates of the **yield of these high-energy neutrons** † indicate an intensity about 10^{-3} **that of the giant-resonance neutrons**. These neutrons have a broad angular distribution which is almost isotropic and so must be considered in the wide-angle shielding.

High-energy neutrons of the 100-Mev (or higher) range come also from high-energy proton accelerators such as the synchrocyclotrons and proton synchrotrons. They are formed in charge-exchange and in direct-collision processes by the primary protons, with a cross section which is approximately the geometrical cross section of the nucleons in the absorber. Because of momentum conservation in the interactions, the angular distribution is more forward than for photoneutrons; the higher-energy neutrons are more strongly forward and lower-energy neutrons are emitted at wider angles. The half angle of the forward cone including 50 per cent of the neutron intensity is about 10° for incident protons of 350 Mev energy, for example.

Absorption measurements of this high-energy neutron component are scarce and show differing results depending on neutron energy and on the measuring techniques used. The absorption coefficient is a function of energy, varying approximately with $E^{1/2}$, but theoretical interpretation is vague. The cross section for absorption is close to that for protons of the same energy, suggesting a common nucleonic-interaction process. At around 100 Mev energy the best-accepted value ‡ of the absorption mean free path is 14 in. of ordinary concrete, leading to a **half-intensity thickness of 10 in. of concrete** or an **absorption coefficient of 0.069 per inch**. For high-intensity accelerators where space is at a premium, the equivalent density of high-density concrete is generally used for shielding.

* *Phys. Rev.*, vol. 84, p. 43, 1951.

† Chodorow, Ginzton, Hansen, Kyhl, Neal, and Panofsky, *Rev. Sci. Instr.*, vol. 26, pp. 134-204, 1955.

‡ Williams, Cambridge Electron Accelerator Shielding Study, CEA-10, Harvard University, Cambridge, Mass., Aug. 31, 1956.

Mesons are produced in neutron and proton impacts with other nuclei, at energies above the meson threshold of about 150 Mev. The yield of mesons and their energy distribution do not depend significantly on target material, since the production process is essentially a nucleon-nucleon interaction. The yield of mesons produced by $1\text{ }\mu\text{a}$ of 450-Mev protons in the University of Chicago synchrocyclotron is about 4×10^{12} mesons/sec. Meson energy extends up to over 200 Mev, with maximum intensity in the region around 100 Mev. The number is about the same as the number of high-energy neutrons.

Interaction cross sections of mesons in matter have been measured in several laboratories. They show a variation with meson energy, having a strong resonance at about 250 Mev and another broad resonance at 1.0 Bev. The average interaction cross section is essentially the same as for high-energy protons or neutrons; so for shielding purposes they can be lumped with the other nucleonic components. It might be noted, however, that ionization losses for the charged particles (protons and charged mesons) absorb a considerable fraction of the energy. The neutron component is dominant and should be used for shielding calculations. The mean free paths for such fast heavy particles in the energy range above 100 Mev can be taken as 14 in. of concrete, for which the half-value layer is 10 in.

Induced radioactivity induced in the chamber or surrounding materials, by the beam or by secondary radiations, is one of the major hazards in accelerator operations. The metal chamber and electrodes of a cyclotron, for example, can become so radioactive that heavy gloves must be worn in maintenance operations, and exposure time for each person must be limited to a few minutes a day. Much can be done in design to minimize the intensity, such as by lining the D's with graphite in which only short-lived activities are induced. Remote-handling techniques may also be developed for target changing, cathode replacement, and other routine operations, and they can be scheduled following long shutdown periods. Nevertheless, personnel exposure to the radioactivity of beta and gamma rays is in most well-shielded cyclotron laboratories the dominant radiation dosage.

The activities cover a wide range in half-lives, coming from a variety of radioactive isotopes as for fission products, and have a similar composite-decay curve. The **induced radioactivity decay rate** measured at a time T after the machine is turned off shows a half-life of approximately T . This rapid initial decay can be used to advantage in scheduling repairs. The composite spectrum of beta and gamma rays is quite similar to the fission spectrum, and shielding requirements are essentially the same.

Very high energy accelerators have much lower dose rates of induced radioactivity; because of the penetrating character of primary and secondary radiations, the activity is produced deep in the shield. The dose rate comes primarily from low-energy degraded radiations such as slow neutrons and is roughly proportional to the power of the beam.

SHIELDING ARRANGEMENTS

Accelerator shielding must be designed with due regard to several factors other than radiation protection. It cannot be as compact as for a pile, since space around the machine is needed for electronic maintenance, for withdrawing the chamber,

for vacuum-leak hunting, for target installation, etc. It must allow for a varied and continuously modified research program. Final beam energy or intensity can seldom be guaranteed in advance; so shields should be capable of expansion to meet growing needs. The first accelerator of a new type has usually been built without any installed shielding. Then, after machine development brought rising intensities and showed the need, the necessary shielding was added, layer after layer. The early installations are therefore not satisfactory models of shield design. With growing experience suitably designed shielded rooms have been planned for later-model machines. However, even in these instances the shields have often been modified to accommodate new requirements of the research program. We must conclude that few if any accelerator-shielding systems are completely satisfactory or adequate. Accelerator-shield design is still a developing art which requires both experience and imagination.

Space and distance are the first considerations in planning accelerator shielding. For low-energy accelerators the radiations are frequently isotropic, and intensity in any direction is approximately inversely proportional to the square of the distance from the target. The distance of personnel from the target should be made as great as possible to take advantage of this reduction in intensity and thus to reduce the required thickness of shielding barriers. Even with an isotropic source the target is frequently shielded locally by the housing or by the accelerator structure; so the emergent radiations may be partially collimated. Furthermore, scattering of radiations from the target housing or by the surrounding walls may distort the isotropic distribution. For economy in construction the shield must be graded in thickness in different directions to accommodate the anticipated radiation dose rates. This is even more significant for high-energy accelerators for which the dominant radiations are strongly collimated. Heavy shields will be needed in the direction of the forward beam from the target, while much less absorber is required on the sides or overhead. The control console and other personnel-occupied areas can be located away from the forward-beam direction.

Radiation dose rate should be estimated (or preferably measured) at the location which personnel will occupy, and the required attenuation factor should be determined by comparison of the unshielded intensity with the desired final intensity based on the chosen safe dose rate. Shield thickness can then be computed from the attenuation coefficient of the chosen material. Protective barriers should be located as close to the accelerator target as possible, to reduce total weight of shielding. Where space is at a premium it may be desirable to use more expensive high-density materials. Lead is commonly used for gamma-ray shielding when it can be located close to the source. High-density concrete loaded with iron ore or steel scrap is preferable for room-wall shielding, where structural strength is also a factor. When distance is not a limitation ordinary concrete or an earth fill can be used.

The material chosen for shielding depends on the **dominant radiation** from the particular accelerator. For cyclotrons the mixture of neutrons and gamma rays is quite similar to that from an atomic pile, and similar materials can be used. Electron accelerators require shielding primarily for high-energy X-rays, and chiefly in the forward-beam direction; lead is the most efficient absorber. Synchrocyclotrons and proton synchrotrons produce high-energy primary protons and secondary neutrons, which are strongly directional and highly penetrating; the major shielding

is in the direction of the primary beams, and high-density concrete is the basic material. The chief component in the degraded radiation at large angles from these high-energy proton accelerators is high-energy neutrons for which the overhead shielding can be ordinary concrete.

The shielding requirements for accelerators are summarized in Table 6-12. A

Table 6-12. Shielding Requirements for Accelerators

Accelerator	Particle	Energy beam	Dominant radiation	Absorber	Absorption coefficient	Typical shield	
						Forward beam	Overhead
Voltage multiplier	P	1 Mev 20 μ a	X-rays (ion source)	Pb	0.8 g/cm ²		
Electrostatic generator	P	4 Mev 200 μ a	X-rays (ion source)	Pb or concrete	0.5 g/cm ² 0.08 g/cm ²	2 ft concrete	1 ft concrete
Standard cyclotron	d	16 Mev 0.5 ma	γ -rays + fast neutrons	Concrete	0.23 per in.	4 ft concrete	3 ft concrete
Betatron	e	20 Mev	X-rays	Concrete	0.15 per in.	2 ft concrete	6 in. concrete
Electron linear accelerator	e	50 Mev 1 μ a	X-rays	Concrete	0.14 per in.	3 ft concrete	1 ft concrete
Electron synchrotron	e	330 Mev	X-rays	Concrete	0.14 per in.	3 ft concrete + Pb slot	1 ft concrete
Synchrocyclotron	P	1,000 r/min 450 Mev 1 μ a	High-energy neutrons	Fe + concrete (d = 4.25)	0.12 per in.	6 ft Fe + concrete	5 ft concrete
Proton synchrotron	P	3.0 Bev 10 ¹⁰ /sec	High-energy neutrons	Fe + concrete (d = 4.25)	0.12 per in.	8 ft Fe + concrete	3 ft concrete

typical accelerator of each kind is listed, with basic parameters defining its size, energy, and beam intensity. The dominant radiation in the forward-beam direction is indicated, and the recommended absorber. Then for each case the appropriate absorption coefficient for the radiation is listed. The last two columns give the recommended shield thickness in the forward direction and overhead for an accelerator of the energy and intensity indicated.

The shielding arrangements for a typical accelerator of each kind are described below. These descriptions are qualitative and are intended to illustrate only the basic requirements.

Voltage multiplier; 1-Mev protons, 20- μ a beam. No general shielding is required except room walls, say 6 in. of concrete. Local shielding of the ion source by a few inches of lead may be useful to absorb X-rays from the source, and some protective guards should be installed around the target area to prevent operators from being burned by scattered protons from the beam.

Electrostatic generator, 4-Mev protons or deuterons, vertical mounting, 200- μ a beam entering a basement laboratory. Remote operation of the generator and also of research apparatus in the laboratory is necessary, from a control console say 20 ft from the machine and behind a 2-ft wall of concrete. X-rays from the ion source

will produce more ionization intensity than will neutrons from the target and should be shielded by several inches of lead around the ion source, 1-ft concrete walls around the entire generator, or both. Radioactivity induced in targets requires modest handling precautions, but the general radioactive level in the room is negligible.

Standard cyclotron, 16-Mev deuterons, 0.5-ma resonant beam, 0.1-ma emergent beam. The MIT cyclotron is enclosed in a completely closed vault with 4-ft ordinary concrete walls, a 3-ft concrete roof, and movable 4-ft concrete doors (see Fig. 6-19). A primary consideration is to provide a tight shield with no apertures

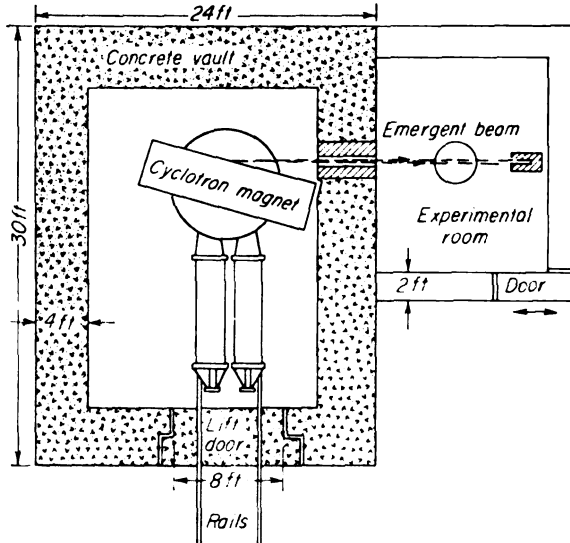


FIG. 6-19. Shielding vault for the MIT 16-Mev cyclotron.

through which slow neutrons can diffuse. Baffled entries are possible but not recommended. If an emergent beam of 10 per cent intensity is brought through a channel in the shielding wall to a target-observation room, this room should be shielded with 3-ft walls and 2-ft roof. Remote operation is essential. When the cyclotron is not operating maintenance personnel can enter the vault for short periods probably not exceeding about 20 min/day because of induced radioactivity inside the vault. Handling of radioactive targets and machine components requires the use of tongs and remote-handling equipment as for a pile hot lab (see Sec. 18).

Betatron, 20-Mev electrons, X-ray beam intensity 100 r/min at 1 m in forward direction, 50 per cent intensity in cone with 5° half angle. Heavy shielding is required in the direction of the forward beam, of concrete or more dense material, with a lighter shield enclosing the betatron. One installation uses a 2-ft concrete front wall having a slot filled with lead brick to define a beam channel; another report describes the use of graphite blocks to define the beam. The choice of absorber rests on the planned use of the beam. Radioactive products are of low intensity.

Linear accelerator, 50-Mev electrons, 1- μ a average beam. The primary radiation is X-rays in the forward direction, for which a 3-ft concrete wall with lead-lined slot is sufficient. Scattered radiation along the length of the accelerator is small, and no side or overhead shielding is required except near the target end. Personnel should be denied access to the target area beyond the shield during operations. If high-voltage klystrons are used for the radiofrequency power source a special problem exists in shielding personnel from the X-rays generated in the klystrons.

Synchrotron, 330-Mev electrons, internal X-ray target, X-ray beam intensity 1,000 r/min at 1 m, half angle of cone 0.4°, emergent electron beam of 6×10^8 per second scattered from target. The MIT synchrotron laboratory arrangements are shown in Fig. 6-20. The side and top shielding required for scattered X-radiation

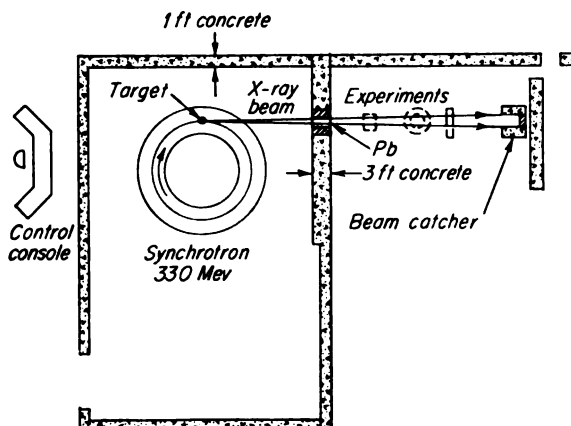


FIG. 6-20. Shielding arrangements of the MIT 330-Mev electron synchrotron.

is about 1 ft of concrete. A 3-ft concrete wall is used in the forward direction with a slot filled with 2 ft of lead brick to define the emergent beam. A special "beam-catcher" shield located beyond the experimental target area is used to absorb the narrow beam crossing the area, and personnel are denied access during operations. No protection for radioactivity is required.

Synchrocyclotron, 450-Mev protons, 1- μ a average current on an internal probe target. These figures describe the University of Chicago accelerator, one of the highest-energy synchrocyclotrons in this country. The shielding was planned with the primary purpose of reducing background intensity for experiments in the observing area behind the primary shield and elsewhere in the building, and is far in excess of that required for personnel protection. The arrangement of shields is shown in Fig. 6-21. The primary shield is 12 ft of iron-loaded concrete with the beams emerging through channels in steel plates in the plane of the beam. Overhead shielding to close the vault around the machine consists of 10 ft of ordinary concrete blocks which are removable by the overhead crane. The level of radioactivity is much smaller than in a standard cyclotron; so maintenance handling is not a serious problem.

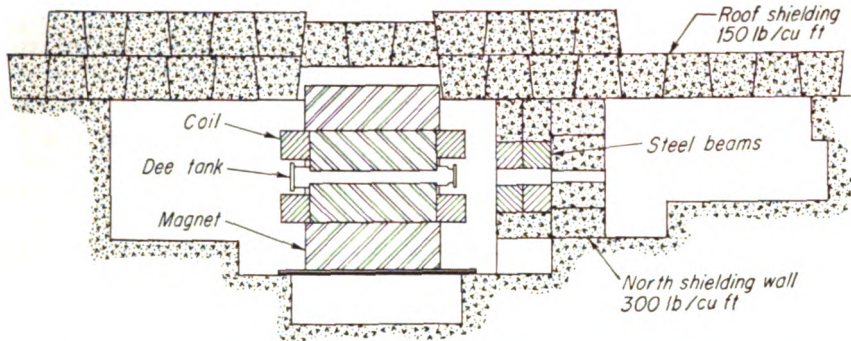


Fig. 6-21. University of Chicago 450-Mev synchrotron shielding arrangements.

Proton synchrotron (cosmotron), 3,000 Mev, 10^{10} protons/sec on internal target, 10^9 per second emergent beam. The original shield was an 8-ft-thick concrete wall around the target sector with a 6-in. layer of high-density concrete bricks at beam level. Degraded scattered radiation coming over the wall required the addition of a 2-ft overhead layer of concrete over the target region. With higher intensities now available this shielding is not adequate. Plans for a new shield completely bridging and covering the ring magnet are now in process. This is an example of a machine in which increased beam intensity has required increased shielding. In general, the shielding required in the direction of the beam is heavier but the overhead shielding is lighter than for lower-energy higher-power machines such as synchrocyclotrons. No radioactivity problem exists.

ULTRA-HIGH-ENERGY ACCELERATORS

The new superenergy accelerators now under construction have become possible through the use of alternating-gradient magnetic fields to provide **strong magnetic focusing**. With a reduced amplitude of transverse beam oscillations the accelerating chambers and the size of the ring magnets to produce the fields are much smaller. It becomes economically practical to design for much larger orbit radii and much higher energy. The **alternating-gradient synchrotron** under construction at Brookhaven is for 25- to 30-Bev protons, ten times higher energy than the cosmotron. Pulse-repetition rate and beam intensity will be much the same. A similar 25- to 30-Bev proton accelerator is being built at Geneva in the C.E.R.N. laboratory. Plans have been announced from the U.S.S.R. for a 50-Bev machine. Meanwhile, construction has started on a 6-Bev **electron synchrotron** using strong focusing magnets at Cambridge, Mass., sponsored by Harvard University and MIT.

These new accelerators enter a new energy range for which the only available information on interaction cross sections and shielding requirements comes from cosmic-ray studies. In all cases the basic shielding plan is to locate the ring magnet in a below-ground circular tunnel. Emergent beams will be brought through horizontal channels in thick shielding walls (12 to 30 ft of high-density concrete) into an experimental observation area.

New shielding problems arise associated with the penetration of the high-energy primary and secondary radiations. Fast protons, neutrons, and π mesons will have up to ten times the energy coming from present accelerators and will require unusually thick shielding in the horizontal plane. These high-energy heavy particles develop nucleonic showers, degrading into a large number of lower-energy particles. Gamma rays from the electron accelerators will develop an electron-photon cascade with a small angular spread. The π mesons decay into μ mesons in flight; μ mesons have extremely small interaction cross sections (10^{-4} to 10^{-5} of those for π mesons) and are absorbed almost solely through ionization, at the minimum ionization rate for velocity-of-light particles of 2 Mev loss per g/cm² of absorber. Other strange and rare particles will be produced, such as K mesons, hyperons, and negative protons, but with much smaller intensities. Shielding calculations suggest that the dominant component will be neutrons of about 100 Mev energy, produced in nucleonic showers or by the photoneutron process; these neutrons have an almost isotropic distribution and will set the requirements for overhead shielding. It is believed that the planned underground laboratories will provide adequate shielding, but our knowledge of the properties of such high-energy radiations is far from complete.

ELECTRICAL SOURCES OF X-RADIATION

Thomas H. Rogers

General References

- Compton and Allison, "X-rays in Theory and Experiment," 2d ed., Van Nostrand, 1935.
- Weyl, Warren, and O'Neill, "Radiologic Physics," Charles C Thomas, 1941.
- Glasser, Otto (ed.), "Medical Physics," vol. 1, pp. 1360-1364, 1391-1401, 1944; vol. 11, pp. 1-20, 1950, Year Book Publishers.
- Clauser, H. R., "Practical Radiography for Industry," Reinhold, 1952.
- Klug and Alexander, "X-ray Diffraction Procedures," Wiley, 1954.
- Clark, G. L., "Applied X-rays," 4th ed., McGraw-Hill, 1955.

X-rays are a form of electromagnetic radiation produced whenever electrons, having been accelerated to high energy in an evacuated chamber by an electric field, are slowed down or stopped by collision with atoms of matter lying in their paths. The electron energy required to produce X-rays has no sharply defined lower limit, but 10^2 volts may be taken as the lowest voltage value having any practical significance. In most cases the voltage E applied to the electrons in the accelerating field will be greater than 10^4 volts. The upper limit is set only by the energy range of modern electron accelerators. The range of electron energies in present use for the production of X-rays is from 10^4 to 10^8 volts.

The **wavelengths of X-rays** so generated constitute a continuous spectrum extending from the ultraviolet region down to a minimum wavelength $\lambda_0 = 12,354/E \times 10^{-8}$ cm, corresponding to the stopping of an electron in a single collision. The distribution of wavelengths within the continuous spectrum is based on the probability aspects of partial and total stoppages by collisions in the target mass. Maximum intensity occurs at approximately $\frac{3}{2}\lambda_0$. Superimposed on the continuous spectrum may be "lines" of intensity maxima whose wavelengths are characteristic of the elements constituting the target material. To excite a group of lines in the **characteristic spectrum** of a given element, the energy of the bombarding electrons must equal or exceed the value corresponding to the shortest wavelength in that group. A typical X-ray spectrum with its continuous and characteristic components is shown in Fig. 6-22.

Vacuum tubes designed specifically for the production of X-rays are known as **X-ray tubes**. An electrical apparatus designed specifically to generate and control electrical energy for application to X-ray tubes is called an X-ray generator.

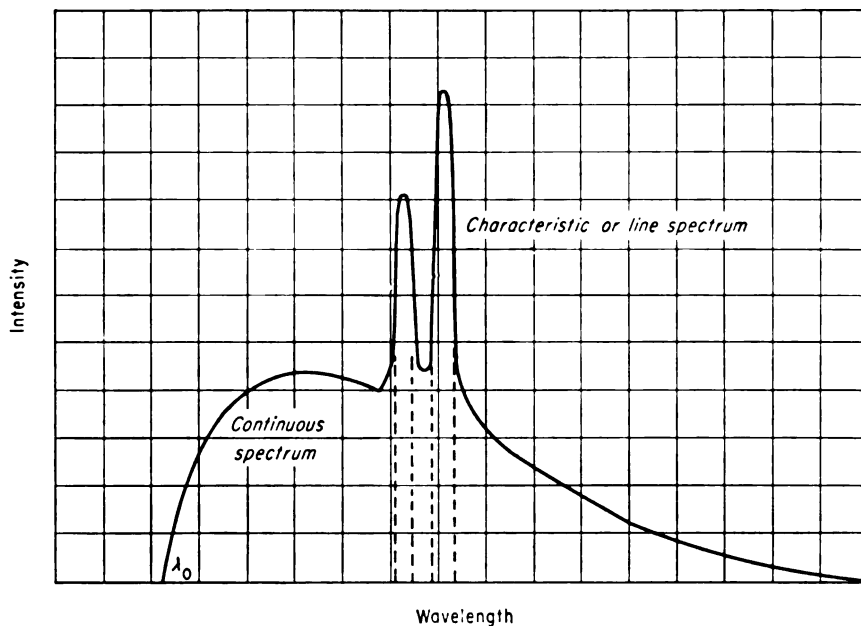


FIG. 6-22. Diagram of a continuous X-ray spectrum with characteristic "lines."

X-RAY TUBES

Designed for many different applications, X-ray tubes differ widely as to structural detail, as illustrated in Figs. 6-23 and 6-30. Modern X-ray tubes utilize high vacuum as the voltage-insulating medium within which the anode and cathode structures are mounted and the electron beam is accelerated. The residual gas pressure in such tubes is usually less than 10^{-6} mm Hg. Early X-ray tubes depended on the ionization of the residual gas to produce the electron beam by ion bombardment of the cathode and required gas pressures about 1,000 times higher. The **anode** incorporates a **target** of suitable metal upon which the electron beam impinges and wherein the X-rays are generated. Tungsten is the most commonly used target material, both because of its high atomic number which makes for efficiency in X-ray production and because of its favorable thermal properties of high melting point, high conductivity, and low vapor pressure. Other metals are used when the specific applications require characteristic spectra, as in X-ray diffraction studies. The **cathode** of an X-ray tube includes the source of electrons, generally an incandescent tungsten **filament**, and an associated beam-forming structure which focuses the electron stream on the target. The area on the target bombarded by electrons which thus becomes the source of X-rays is called the **focal spot**. The size and shape of the focal spot are designed for specific applications and are controlled primarily by the beam-forming structure of the cathode. The bombarded area must be large enough to absorb safely the thermal energy delivered by the electron beam, but not so large as to produce excessive penumbra when the X-rays are re-

quired from an approximate "point source," as in diagnostic radiology. A cooling system is provided for transfer of heat from the target to the exterior of the tube and its ultimate disposition. Finally, the tube incorporates a structure which supports and maintains the cathode and target in proper spatial relationship and which insulates these elements from each other at the operating voltage. Usually the envelope of the tube is used as the insulating structure as well as the container of the evacuated space. The envelope may be immersed in an insulating medium such as oil of greater dielectric strength than air, which in turn is contained in a grounded metal enclosure which provides protection against electric shock and shields off unwanted portions of the X-radiation. Such units are generally spoken of as having a **protective tube housing** (see NBS Handbook 60).

Medical and dental diagnostic X-rays are usually generated at voltages between 40 and 150 kv.* Dental and lightweight (portable) medical units are usually limited to 90 kv or less. The usual maximum range for diagnostic equipment is 125 kv. Diagnostic technique may take the form of **radiography**, wherein X-ray shadow pictures are produced on photographic film, or of **fluoroscopy**, wherein a visible light image is produced by X-rays on a fluorescent screen. Both techniques require small focal spots for sharply defined images. Radiography requires relatively high instantaneous energy application to the tube so as to produce the radiograph in a period sufficiently short to prevent blurring due to motion. Fluoroscopy requires applications of energy to the tube for a period sufficiently long to permit adequate examination. Level of energy is a compromise between that required for adequate image brightness and minimized patient dosage.

Tubes designed for medical diagnosis are generally usable for both radiography and fluoroscopy. In their design, particular attention is paid to the provision of the minimum focal-spot size consistent with the electrical loading factors required for the intended service, and to rapid removal of heat so that limitations on repetitive operation are minimized. Simplest types employ a tungsten target embedded in a copper member, called the anode, having a portion extending to the exterior of the tube envelope. The heat generated at the focal spot is transmitted into the copper and conducted thereby to the exterior and transferred to the surrounding medium. If the surrounding medium is air, the tube is called an **air-cooled X-ray tube**. More

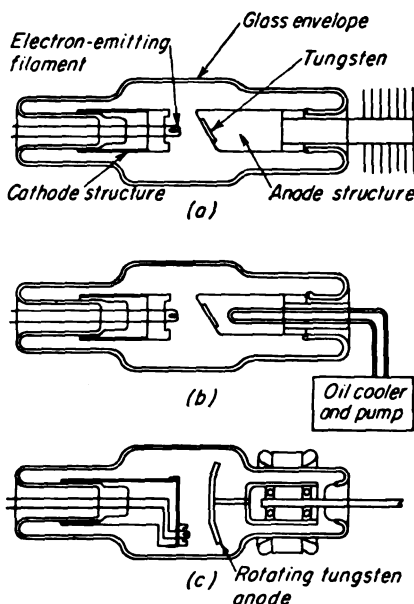


FIG. 6-23. Schematic diagram of X-ray tubes. (a) Air-cooled tube. (b) Forced-oil-cooled tube. (c) Rotating-anode tube.

* In X-ray practice, voltage values usually refer to the peak or crest value in the case of pulsating or alternating voltage waveforms; frequently written as pkv or kvp.

often, in modern practice, the tube is immersed in a housing filled with insulating oil; such a tube is called an **oil-immersed tube**. Heat is transferred to and carried away by circulation of the oil. Sometimes another insulating material, such as compressed gas, is used. Allowable minimum focal-spot size in such tubes is governed by rate of heat transmission through the tungsten target to the copper backing.

Tubes designed to provide the smallest possible focal spot for a given loading capacity for short-exposure radiography employ a **rotating anode**, wherein the target is a relatively large-diameter disk, made to rotate about its central axis at about 3,000 rpm. The focal spot is near the edge of the rotating disk, remaining stationary with respect to the exterior of the tube. The heat generated is swept rapidly out of the bombarded area by the motion of the target, and thus a much smaller focal spot may be utilized without becoming overheated. Generally, in such tubes, the target is not backed with copper; the heat is spread throughout the tungsten disk and radiated therefrom with satisfactory efficiency because of the area available for radiation and the high temperature permissible with the all-tungsten target assembly. The rotating anode assembly is mounted on a special ball-bearing system specially designed for operation in vacuum, with a nonvolatile metallic lubricant. It is driven by induction-motor action, an external stator providing the magnetic field.

Diagnostic tubes produce radiation dosages varying very widely in amount depending on the particular radiographic or fluoroscopic technique employed. Use of suitable aluminum filter (approximately 2 mm) in the beam is usually recommended to reduce the incident dose to the patient (see Secs. 3 and 13). Radiation intensity emanating from the tube outside the useful beam, in the case of rayproof diagnostic tubes conforming to accepted standards, does not exceed 0.1 r/hr at a distance of 1 m when operating at maximum rated voltage and maximum average current allowable as a continuous rating.

TUBES FOR MEDICAL THERAPY

A very wide range of voltage and current is employed in the generation of X-rays for medical therapy, depending on the nature of the therapeutic problem involved. Such therapy is broadly divided into two classes: **superficial therapy**, pertaining to skin and subcutaneous disorders; and **deep therapy**, pertaining to more deeply located treatment areas. For superficial therapy, voltages from 6 to 150 kv are employed; for deep therapy, voltages range from 180 kv to 50 Mev.

Superficial-therapy X-ray tubes for the voltage range of 30 to 150 kv are generally similar to those employed for diagnostic techniques. Dosage requirements are usually greater and exposure times are not required to be limited to short duration. Hence, over-all cooling rather than focal-spot considerations is the primary factor governing design, and rotating anode tubes are seldom used for this application. For some superficially located conditions, extremely long-wavelength radiation of low penetration characteristics is required. Tubes for this class of therapy are operated at voltages between 6 and 50 kv and are frequently provided with an X-ray exit window of metallic beryllium in the tube envelope. Such **beryllium-window tubes** permit the emission of the longer-wavelength portion of the spectrum which is normally absorbed by other types of X-ray window. The dosage level at the

outer surface of the window in such tubes may be over 1,000,000 r/min, and hence special safety precautions are necessary in their use. However, the high absorption coefficient makes the shielding of such radiation relatively easy.

X-rays generated at voltages from 6 to 15 kv are often called **grenz rays**. Modern grenz-ray tubes are usually equipped with beryllium windows.

For the treatment of lesions located within bodily cavities, special X-ray tubes are designed for insertion into such cavities so that the X-ray window of the tube can be brought into very close proximity or actual contact with the lesion. Such tubes, usually called **contact-therapy X-ray tubes**, generally operate at voltages between 40 and 60 kv.

Deep-therapy X-ray tubes occur in a variety of forms, the most common of which is the oil-cooled type used in units operating at 200 to 300 kv. This service requires prolonged periods of loading at relatively high power (approximately 4 kw), so that dissipation of this large amount of heat from the anode becomes the primary factor in design. The tungsten target is embedded in a copper anode, as in the diagnostic tubes described above, but instead of being dependent on conduction of heat to the exterior for cooling, the copper anode is made hollow and insulating oil is pumped through it as a coolant to carry the heat away as rapidly as it is generated. Such tubes are usually mounted in a shockproof, rayproof enclosure and insulated therein by the same oil used as the coolant. In operation the anode may be at a potential of 100 to 150 kv from ground; hence, good insulating properties are required.

Some deep-therapy tubes, which are generally cooled by water circulating through the anode structure, are designed for operation with the anode at ground potential. These tubes have the advantage of utilizing the superior cooling properties of water, and also usually have the target located at the extreme end of the envelope, providing certain geometric advantages in application.

There are physical advantages, having to do with the absorption coefficient of the radiation, in employing extremely high voltages (1,000 kv and above) for deep therapy. Tubes for use in this range are often called **supervoltage X-ray tubes**. Their construction is usually of the grounded-anode type described above, and also is usually of the multistage type, whereby the envelope is made up of a number of interconnected sections in which the electron beam is subjected to successive accelerating fields as it passes through them in turn. Thus each section is required to insulate only a fraction of the total voltage involved. A "tapped" voltage source is required to energize such tubes. Multistage construction is regularly employed for voltages of 1,000 kv and above, and in some cases for lower voltages down to 250 kv.

In the range of 5 to 100 Mev, X-rays are generated by the **betatron**, a device which accelerates electrons to high energies in a circular path through multiple revolutions by means of magnetic induction. The X-ray tube in this case is in the form of a hollow toroid, often called a **donut**. The donut contains an electron "injector," similar to the cathode in a conventional X-ray tube, which injects electrons at an initial energy of about 50 kv. A magnetic field causes them to follow the circular path, and linkage of the circular path with an increasing magnetic field provides the accelerating induction. At a time of maximum velocity, the electrons strike a target within the donut and produce X-rays by the usual collision mechanism. The usual target-cooling problems are essentially absent because the effi-

ciency of X-ray generation at such high energy levels is such that little heat generation takes place at the required X-ray output.

X-ray units used for medical therapy usually operate at dose rates, suitable for treatment routines, ranging from approximately 30 to 100 r/min at the treatment area (usually from 20 to 100 cm from the target). "Quality" (wavelength and homogeneity) of the radiation is controlled by inserting in the beam "filters" of aluminum, copper, tin, lead, etc., which greatly reduce the primary-beam intensity. Precautions to avoid inadvertent treatment without prescribed filters are essential. The emission of radiation outside the useful beam in therapy tubes is usually limited to a maximum of 1.0 r/hr. Personnel protection must take into account scattering from the patient and other objects in the beam as well as "leakage" radiation from the X-ray tube housing.

TUBES FOR INDUSTRIAL APPLICATIONS

Industrial applications of X-rays occur in a number of categories, the most common of which is inspection by **radiography**. As in medical radiography, X-ray shadow pictures of objects to be inspected for internal defects or structural details are produced on photographic film. Sharp images require small focal spots. For industrial subjects, usually motion is not a factor; so that long exposures requiring relatively low power levels may be used, permitting focal spots as small as may be desired. Voltage employed is governed by the nature and thickness of the material to be penetrated and may range from a very few kilovolts to several million volts. The design of tubes for **industrial radiography** is generally similar to that of tubes for medical therapy for corresponding voltage ranges, with particular attention to optimizing focal-spot size, which is generally considerably smaller in tubes for radiography with corresponding reduction in power rating. The betatron is frequently used for radiography in the 5- to 25-Mev range, for examination of heavy metal objects up to 20 in. thick.

A special technique called **microsecond radiography** is employed to study rapidly moving objects (projectiles, mechanisms, etc.). In tubes for this technique, the cathode contains a small gap between electrodes across which an arc is caused to fire by a suitable trigger-voltage pulse. This arc becomes a copious source of electrons to convey an instantaneous discharge current to the anode, producing a very short highly intense burst of X-rays. The discharge current is supplied by a charged condenser bank at voltages up to 360 kv. Capacitance and other circuit constants determine the duration of the discharge, which usually is in the order of 1 μ sec. The target is a block of tungsten, cooled primarily by radiation and by the formation of tungsten vapor. Life of such tubes is necessarily quite short, a few hundred discharges evaporating sufficient tungsten to render the tube inoperable.

Fluoroscopy is also used for industrial inspection. Tubes for this purpose entail essentially the same physical, electrical, and thermal characteristics as for radiography, up to about the 250-kv voltage level. Fluoroscopic screens become increasingly inefficient at the higher voltages, and hence attempts to use fluoroscopy above this level are uncommon. Special **rotating anode tubes** capable of continuous operation at relatively high power with extremely small focal spots have been developed, which permit fluoroscopy to be utilized advantageously for many crucial

inspections requiring detection of extremely minute defects. These tubes employ very large diameter tungsten disk targets in the manner described above (see rotating anode), permitting dissipation of heat by radiation at a continuous rate of several kilowatts.

Other miscellaneous uses of fluoroscopy include **X-ray shoe fitting**, inspection of automobile tires in place, and **anticrime applications** such as detection of illegal articles in postal packages, smuggled or pilfered articles in possession of personnel, etc. Such applications require special care to avoid undue radiation hazards.

X-ray diffraction, used for microstructure analysis, constitutes another important industrial application. This technique generally makes use of a monochromatic, or single-wavelength, X-ray beam. Tubes for this purpose usually employ a special target material whose characteristic spectrum provides a "line" of the required wavelength (see **characteristic spectrum**), the other wavelengths being essentially eliminated by a suitable filter or other form of "monochromator." In most cases, these tubes are operated with water cooling with the anode at ground potential. The focal spot is usually in the form of an elongated narrow band on the target. Two, three, or four windows for egress of X-ray beams are provided, these windows usually consisting of beryllium, mica, or low-absorption glass, the object being to minimize loss of intensity by absorption in the window. Such tubes are usually operated at 30 to 50 kv, 15 to 20 ma. Target materials most commonly employed are molybdenum, copper, iron, cobalt, and chromium.

X-ray thickness gauging is employed to monitor or control thickness of continuously moving sheet and strip materials, such as steel, copper, aluminum, and plastics, during their manufacture. The operating voltage, dependent on the material and thickness involved, may range from 10 to 150 kv. Tube currents are usually quite low (0.2 to 5 ma). The thickness measurement or comparison is based on the reduction in intensity of the X-ray beam in passing through the material whose absorption characteristics are known. Beam intensity is measured by Geiger or scintillation-counter circuitry and translated into thickness indication or servo-type control signal. Tubes for gauging applications are sometimes required to provide a dual beam, in which case they may be rather similar to tubes used in diffraction apparatus. Otherwise, tubes as used for conventional industrial radiography in the same voltage range are employed in X-ray thickness gauges. Unusual mechanical ruggedness, requiring special structural design, may be demanded in some cases.

X-rays are sometimes employed for the **irradiation of materials** to produce chemical or biological changes, such as partial or complete destruction of microorganisms (sterilization), deinfestation, etc. Special tubes for such purposes have been constructed with large-area beryllium windows and extremely large focal spots, for operation at relatively low voltage and high current (60 kv, 200 ma), for the production of very high dosages of long-wavelength X-rays. Dose rates of the order of 10^7 r/min are produced with such tubes. The low penetration of such rays limits their usefulness to surface and thin-film applications. Other irradiation schemes employ electron beams which are produced and accelerated as in an X-ray tube but allowed to pass through a suitable window to the exterior of the tube and impinge on the material to be irradiated. Such beams must be accelerated by at least 500,000 volts (usually more) in order to provide practical results. A certain amount of

highly penetrating X-radiation is produced in the process, imposing personnel-protection problems equivalent to those of X-ray equipment installations of the same voltage level.

ELECTRIC CIRCUITS FOR ENERGIZING X-RAY TUBES

The most commonly employed device for generating high voltage for application to X-ray tubes is a high-ratio step-up transformer, supplied with alternating voltage from power lines at conventional distribution power and frequency. Auxiliary apparatus, called the **X-ray control**, provides means for varying the input and correspondingly the output voltage of the transformer at will over the required range. Some form of multitapped autotransformer is usually employed for this purpose. The control also makes provisions for supplying and regulating the heating current to the X-ray tube filament so as to control its electron emission to the desired values. Other control functions are to initiate and terminate exposures and govern their duration, regulate heating current supplied to rectifier-tube filaments, and give indications by meters or otherwise of the various current and voltage values established by the various control settings.

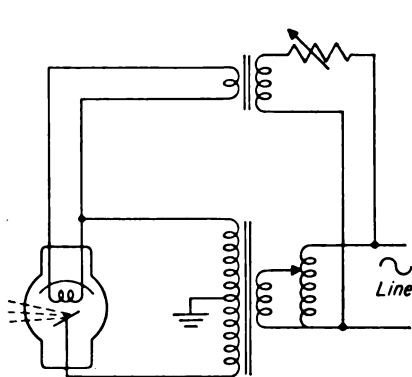


FIG. 6-24. Diagram of self-rectified generator.

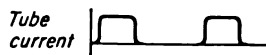
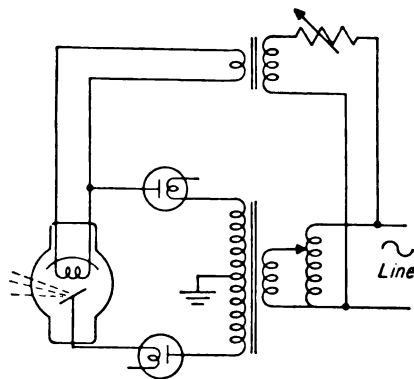


FIG. 6-25. Diagram of half-wave rectified generator.

A **self-rectified X-ray generator** (Fig. 6-24) is one in which the X-ray tube is connected directly across the high-voltage transformer output terminals, so that alternating voltage is supplied to the tube. Current through the tube is unidirectional by virtue of the ability of the tube to function as a rectifier. This mode of operation requires that loading of the anode be limited to a value below the point at which its temperature will allow electron emission on the reverse polarity of the impressed

voltage. By insertion of a high-voltage rectifying tube in series with the X-ray tube, reversal of current is prevented and this limitation on anode loading is removed. Such a unit, with one or two rectifiers, or valve tubes, in series with the X-ray tube, is called a **half-wave rectified X-ray generator** (Fig. 6-25), since each alternate half of the voltage wave is suppressed. In a **full-wave rectified X-ray generator** (Fig. 6-26), four valve tubes are employed in a bridge circuit so as to supply both half cycles of voltage to the X-ray tube with unidirectional polarity. Figures 6-24 to 6-26 show in schematic form the circuit arrangements of transformer, valve tubes, and X-ray tube in the three classes of rectification. When very large currents in the X-ray tube are required, a three-phase voltage supply with full-wave rectification, employing six valve tubes, is sometimes used.

The various types of rectified circuits described above are all commonly used in medical diagnostic X-ray equipments, small, light-duty, and portable units usually being self-rectified. High-current radiographic units are usually full-wave, either single-phase or three-phase. A system for extremely high current, extremely short duration exposures (0.001 to 0.003 sec) employs energy-storage capacitors

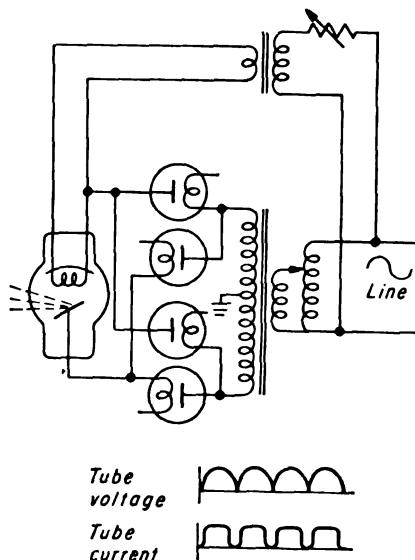


FIG. 6-26. Diagram of full-wave rectified generator.

across the rectified high-voltage source to supply the high-current pulses under control of a high-voltage electronic switching tube of the grid-controlled type.

X-ray-therapy generators may employ any of the forms of rectification described above. A so-called **constant-potential circuit** (Fig. 6-27) is often used in deep-therapy units for operation at 200 to 250 kv. In one such circuit, the transformer, through rectifier tubes, charges separate condensers on respective half cycles, these condensers being arranged to discharge in series continuously through the X-ray tube. In this

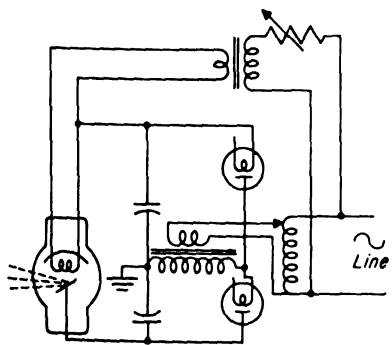


FIG. 6-27. Diagram of constant-potential generator.

way the output voltage of the transformer is doubled and the voltage to the X-ray tube is approximately constant, resulting in increased X-ray output.

For voltages ranging from about 200 kv to 2 Mev, the so-called **resonant-trans-**

former system of generating high voltage (Fig. 6-28) has been employed in a number of instances. In this system, the inductance represented by the transformer secondary is "tuned" to the operating frequency by means of the capacitance of the secondary and its high-voltage terminal. This tuned circuit is designed with a high Q , so that a high circulating current and a correspondingly high terminal voltage is obtained. By using an elevated frequency, such as 1,000 to 2,000 cps, extremely compact and lightweight units can be designed for relatively high voltages. These units are in the self-rectified category, since the X-ray tube is subjected to alternating voltage.

For voltages in the range of 1 to 6 Mev, the **Van de Graaff electrostatic generator** (Fig. 6-29) has been employed with great success. In this type of unit, negative

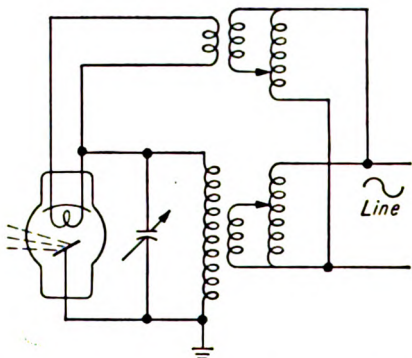


FIG. 6-28. Diagram of resonant transformer type of generator.

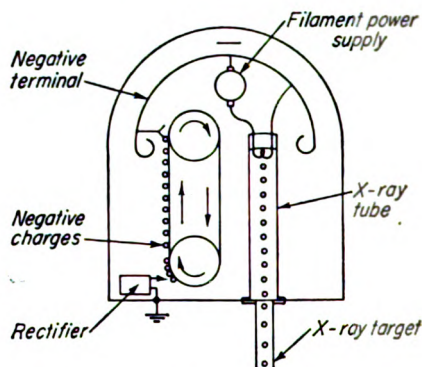


FIG. 6-29. Diagram of Van de Graaff electrostatic generator.

electric charges are sprayed on a moving insulating belt at a relatively low voltage (approximately 50 kv). These charges are mechanically transported to the insulated high-voltage terminal of the machine where they are removed and accumulated. The voltage is maintained at any desired constant value by balancing at that voltage the electrons flowing from the terminal via the X-ray tube with the electrons continuously replenished by the charge-carrying belt. Currents ranging from 0.25 to 4.0 ma can be delivered to an electron or positive-ion accelerator tube at voltages up to 6 Mev by this type of device. Over 10-kw output electron power has been produced by Van de Graaff electrostatic accelerators for radiation processing. Another type of electrostatic generator employing a rotating glass cylinder to transport charges has been developed in France which operates in a range of 100 to 600 kv and is being employed as an X-ray generator.

The **betatron** (Fig. 6-30) employs magnetic induction rather than high voltage to accelerate electrons to high energies; this principle is best suited to energy ranges from 5 to 100 Mev. The electrons make hundreds of thousands of revolutions in the circular path provided in the **donut** tube described previously (see donut). The betatron machine is essentially a large magnetic core, energized by alternating current similarly to a transformer, in which the "donut" acts as a secondary winding and the electron revolutions are analogous to the secondary transformer turns. The

apparatus also includes a large capacitor bank which helps supply the reactive kva required to energize the betatron magnet. Control of betatron injection and the orbit-expanding pulse must also be provided by appropriate circuitry.

Still another device employed for acceleration of electrons and other charged particles is the **linear accelerator** (see Sec. 6). This device utilizes extremely high frequency (microwave) energy, in a series of resonant cavities or a waveguide, to set up instantaneous high electric fields which move along in synchronism with the electrons of a beam injected into the system. For X-ray purposes, such accelerators are in use at energy levels from 6 to 50 Mev. The high-frequency energy is supplied by radar-type oscillators, employing klystrons, magnetrons, or other similar devices, in very short but intense bursts or pulses. Average beam-current values are usually well under 100 μ a, but because of the high electron energy the X-ray yield may be quite high. For radiation processing, microwave linear accelerators with over 5 kw of output electron power are feasible.

All the types of generators described above are used for industrial radiography, the required penetrating power determining the selection of the voltage range. All types are also being used in certain instances for deep therapy. All the generator circuits described for voltage ranges above 1 Mev are also used for the production of electron beams for direct application externally of the tube.

Because of the wide range of voltage and power values employed in X-ray applications, personnel-protection problems cannot be generalized. Adequate personnel practices as well as suitable barriers for direct and scattered radiation, commensurate with the individual installation, as determined by standards for X-ray protection, must be employed.

INCIDENTAL X-RAY SOURCES

Certain electrical devices other than X-ray tubes may also become sources of X-radiation. In general, any vacuum tube or gas-discharge tube in which electrons are subjected to electrical-potential drop of 20 kv or more should be regarded as a possible source of X-rays. Usually the amount of radiation escaping from any such device other than an X-ray tube will be negligibly small if the voltage is under 30 kv. For higher voltages, definite steps to ascertain radiation levels in pertinent surrounding areas should be taken. Devices falling in the category covered by this paragraph include the items discussed below.

Devices using **cathode-ray tubes** should be suspect. Home television receivers do not give off significant amounts of X-radiation when employing conventional metal or glass direct-viewing picture tubes. Projection-type tubes may operate at

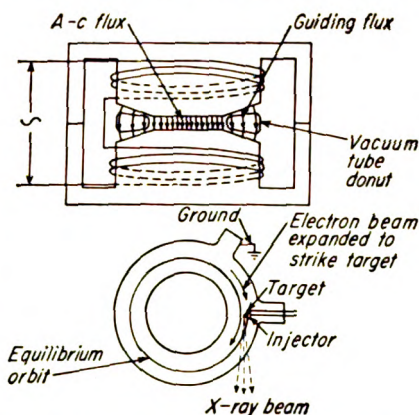


FIG. 6-30. Diagram of betatron principle.

voltages sufficient to produce measurable amounts, which should be suppressed by suitable shielding in the construction of the set. The same considerations apply to cathode-ray **oscilloscopes**, **radarscopes**, etc. The designer will be able readily to provide adequate shielding, since the intensity and penetration of radiation from such sources are not great.

Electron microscopes, operating at voltages from 50 to 100 kv, give rise to fairly penetrating X-rays, but because of the extremely small currents involved the intensity of radiation is small. However, shielding for the protection of the operator should be considered in the design of these instruments.

Rectifier tubes operating in high-voltage circuits may become X-ray sources because of field currents originating from the anode on the inverse voltage cycle. From X-ray apparatus, electrostatic precipitators, cable-testing sets, etc., employing thermionic rectification, such incidental X-radiation may reach appreciable intensity levels, and protective shielding will be required if personnel are to occupy adjacent areas.

High-voltage power tubes operating at voltages above 25 kv may become incidental X-ray sources. **Klystrons** and **magnetrons** operating in high-power radar equipment should be particularly suspect. Incorporation of adequate shielding in the construction of such equipment should be regarded as a design requirement.

X-RAY IMAGE INTENSIFIERS

Devices whose function is to intensify the brightness of the fluoroscopic image produced by X-rays have been developed in recent years, and further developments in this field are continuing. Their need arises from the fact that the brightness of the image on the conventional fluoroscope is so low that visual acuity is likely to be impaired in direct viewing at radiation levels consistent with permissible dosage to a patient. Methods of intensification employed in devices now in regular or experimental use are described below.

The **X-ray image-converter** tube consists of an evacuated envelope containing a fluorescent surface in close proximity to a photoemissive surface. The X-ray image produces a light image which in turn causes electron emission from the photosurface in a pattern corresponding to the light image. This electron image is accelerated and focused by an electrostatic "lens" onto a second fluorescent surface which reproduces the original image as a visible light image of reduced size and increased brightness. Viewing is through a lens system which remagnifies the image to the original size. The accelerating voltage employed with such tubes may be sufficient to generate X-rays, but the construction of such devices is usually such as to eliminate need for further shielding.

Intensification of the brightness of fluoroscopic images has been achieved on an experimental basis by electroluminescent panels, wherein a faint fluorescent image is made to glow more brightly by the application of a transverse electric field. Such devices may in time supplant the conventional fluoroscopic screen.

Still another system being used to enhance the brightness of viewable fluoroscopic images employs electronic amplification in the manner of television reception. The X-ray image is converted into video signals by either of two methods. One method is to produce the usual image on a conventional fluoroscope and view it through a high-

aperture lens with a conventional television camera employing a special type of image orthicon of high sensitivity to low-light levels. The other method employs a special pickup tube having a large-diameter front surface coated interiorly with a photoconductive film sensitive to X-rays. A scanning electron beam converts the X-ray-induced pattern of varying conductivity into video signals which are then amplified and reproduced for viewing by the video receiver. High brightness levels for daylight viewing are readily obtainable on the kinescope screen. Such systems are not presently in extensive use because of cost and need for further refinement in image definition at low signal levels. Additional development is expected to bring about eventual improvement in these factors and increased use of this type of system.

NUCLEAR REACTORS

William M. Breazeale

A nuclear reactor is an apparatus in which nuclear fission may be sustained in a self-supporting chain reaction (see Glossary, Sec. 2). The terms **reactor** and **pile** have been used interchangeably, although the latter is usually reserved for the graphite-moderated natural-uranium reactors.

REACTOR PHYSICS

An outline of certain aspects of nuclear physics which have a direct bearing on reactor operation follows. A self-sustaining chain reactor is possible because the fissioning of the nucleus releases extra neutrons which more than compensate for the loss of the neutron whose capture by the nucleus initiated the fission event. A certain minimum amount of fissionable material, the **critical mass**, must be present in the reactor for the reaction to be self-sustaining. This mass is a function of the particular fissionable material employed, structural materials, geometrical shape, etc.

Neutrons from fission are born with high energies (fast neutrons). An empirical energy relation is *

$$n(E) dE = \sqrt{\frac{2}{\pi e}} \sinh \sqrt{2E} e^{-E} dE \quad (6-11)$$

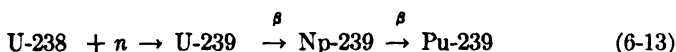
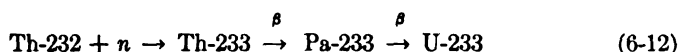
where $n(E)$ is normalized to one neutron and E is the neutron energy in Mev. Shielding calculations are sometimes based on the assumption that all neutrons are released with an energy of 2 Mev.

The probability of neutron capture by the uranium or plutonium nucleus is much greater for slow neutrons than for fast ones, and reactors, with certain exceptions mentioned later, employ **moderators** to slow the neutrons to thermal equilibrium with their surroundings. The neutrons impart their energy to the nuclei of the moderator by a series of elastic collisions. These moderators are light materials that exhibit small **capture cross sections**, i.e., small probability of capturing a neutron. Common moderator materials are light water, heavy water, beryllium, beryllium oxide, and graphite (carbon).

The isotope Uranium-235, which makes up 0.7 per cent of natural uranium, is the only substance occurring in nature which is fissioned by neutrons of thermal

* Watt, B. E., Energy Spectrum of Neutrons from Fissions Induced by Thermal Neutrons, *Los Alamos Sci. Lab. Rept. 718*, December, 1948.

energies. Two artificial isotopes, Uranium-233 and Plutonium-239, which are fissionable with thermal neutrons, can be produced in a reactor from thorium or uranium by the following reactions:



U-233, U-235, and Pu-239 are referred to in the U.S. Atomic Energy Act of 1954 as **special nuclear materials**; thorium and Uranium-238 are defined as **source materials**.

Several determinations of the energy released per fission in U-235 have been made. An energy distribution, taken from Glasstone, follows:

	Mev
K.E. of fission fragments.....	168
K.E. of neutrons.....	5
Fission gammas.....	5
Fission-product betas.....	7
Fission-product gammas.....	6
Neutrinos (not detectable).....	11

These items total 191 Mev per fission. Meem * gives a value of 193 ± 5 Mev per fission from measurements in the Bulk Shielding Facility at Oak Ridge National Laboratory. This indicates that 3.2×10^{10} fissions release 1 watt-sec of energy.

About 15 per cent of the thermal neutrons captured by U-235 lead to the formation of U-236 rather than to fission. Each capture of this type releases a 6.8-Mev photon, and in addition capture gammas from the moderator, coolant, and structural material are present, increasing the total amount of heat that must be removed. It is not clear whether the fission gammas comprise five 1-Mev photons † or two 2.5-Mev photons.‡ To be conservative, shielding calculations should be based on both.

If neutron flux is defined as the number of neutrons per cubic centimeter in a given energy range multiplied by the average velocity of these neutrons, an approximate relation between power released in fission and neutron flux can be written using these values:

$$P \simeq \frac{\sigma N \phi}{3 \times 10^{10}} \quad \text{watts/cm}^3 \quad (6-14)$$

where σ = cross section for fission, barns, at the neutron energy considered

N = No. of fissionable nuclei per cm^3

ϕ = flux in neutrons per sec- cm^2 over the energy range considered

One barn is equal to 10^{-24} cm^2 .

* Meem, J. L., Energy Released for Fission, *Nucleonics*, vol. 12, pp. 5, 62, May, 1954.

† Deutch and Rotblat, Investigation of Fission Gamma Rays, *AEC Rept. AECD-3179*, 1944.

‡ Kinsey, Hanna, and Van Patter, *Can. J. Research*, vol. 26A, p. 79, 1948.

For a reactor in which fissions are caused by thermal neutrons the above becomes approximately

$$P \sim 5 \times 10^{-11} \phi_{av} G \quad \text{watts} \quad (6-15)$$

where G = reactor loading, g U-235

P = reactor power, watts

ϕ_{av} = average flux, neutrons/sec-cm²

The products of fission are radioactive and decay with varying half-lives. A theoretical analysis of the energy available in fission has been carried out by Way and Wigner.* Their relations for the rate of release of gammas and betas after fission are

$$\text{Gammas/sec/fission} = 1.6 \left(\frac{1}{t} \right)^{1.2}$$

$$\text{Betas/sec/fission} = 3.2 \left(\frac{1}{t} \right)^{1.2}$$

where t = time, sec after fission

The average gamma photon energy is 0.7 Mev and average beta energy is 0.4 Mev. A few minutes after shutdown from long-time operation the activity in curies is approximately equal numerically to the previous reactor operating power in watts. An empirical relation, based on Way and Wigner's work, expressing the energy release after shutdown from long-time operation, is

$$P = 0.05 P_0 \left(\frac{1}{T} \right)^{0.3} \quad (6-16)$$

where T = time after shutdown, sec ($10 \text{ sec} < T < 8 \text{ weeks}$)

P = rate of heat release T sec after shutdown

P_0 = reactor operating power

Solid-fuel elements when removed from a reactor after use may remain inside a shield. An approximation to the required thickness of the shield can be computed from Eq. (6-16) with the assumption that the energy is given off in 1-Mev photons. A more accurate computation can be based on values given in Table 6-13.

Reactivity is defined as the fractional change in the number of neutrons in the system per neutron lifetime. The lifetime is the time required for the neutron to slow to thermal equilibrium plus the time it remains at thermal energy before being captured. This lifetime varies from a millisecond for a large graphite-moderated reactor to a few microseconds for a fast breeder (q.v.). If the reactivity is unity the chain reaction is just self-sustaining; if greater than unity the neutron density and the power increase, and if less than unity both decrease.

About 0.75 per cent of the neutrons released during a chain reaction are produced by decay of certain of the radioactive fission products and appear after the fission event.† Five groups of these **delayed neutrons** have been identified, the group having the longest delay developing from a precursor with a half-life of 55 sec. The delayed neutrons immensely simplify the task of controlling a reactor, but in circu-

* Way, K., and E. P. Wigner, *Phys. Rev.*, vol. 73, p. 1318, 1948.

† Glasstone, S., "Principles of Nuclear Reactor Engineering," p. 230, Van Nostrand, 1955.

Table 6-13. High-energy Gamma Rays from Fission Products *

Radioactive species	Half-life †	Fission yield, per cent	Gamma yield, per cent	Energy, Mev
Ru ¹⁰⁶ , Rh ¹⁰⁶	1y, 30s	0.48	2	2.9
Ce ¹⁴⁴ , Pr ¹⁴⁴	290d, 17.5m	5.3	Small	>2.2
			2	2.2
			2	2.6
Eu ¹⁵⁶	15.4d	0.013	60	2.0
Ba ¹⁴⁰ , La ¹⁴⁰	12.8d, 40h	6.1	3.2	2.5
Te ¹³² , I ¹³²	77h, 2.4h	4.5	2.7	2.0
Te ^{131m} , Te ¹³¹	30h, 25m	0.45	21.6	>2.2
I ¹³⁵	6.7h	5.6	1.95	2.4
			4	1.8
Kr ⁸⁸ , Rb ⁸⁸	2.77h, 17.8m	3.1	15	2.8
			19-34	1.9

* From Glasstone, Samuel, "Principles of Nuclear Reactor Engineering," Van Nostrand, 1955.

† y = year, d = day, h = hour, m = minute, s = second.

lating-fuel types they are carried out of the core and induce a certain amount of radioactivity in the exterior piping and heat exchanger.

Some of the fission products have large cross sections for neutron capture. In particular Xenon-135, which results from about 6 per cent of all fissions, has a very large capture cross section for thermal neutrons. The xenon concentration reaches a maximum about 12 hr after the reactor is shut down and then decays. If the reactor is to be started up during the period of large xenon concentration extra fuel must be provided to offset the effect of the parasitic neutron captures. The problem is serious when the neutron flux exceeds 10^{14} neutrons/cm²-sec.

Breeders or converters are the designations given to reactors which produce fissionable material while fuel is being consumed. Each capture of a neutron by a fissionable nucleus results in the release on an average of 2.1 to 2.9 neutrons * (depending on the isotope under consideration and the energy of the incident neutron). Thus in principle a reactor so constructed that the extra neutrons, beyond those required to maintain the chain reaction, are captured in Th-232 or U-238 can produce as much or more fissionable material than it consumes. Actually, because of parasitic neutron capture in fission-product poisons, structural material, and coolant, it is very difficult to attain this objective. If the fissionable material produced is identical with the fuel consumed, the reactor is referred to as a **breeder**; if different, a **converter**. The thorium cycle [Eq. (6-12)] is most efficient with thermal neutrons and the uranium cycle [Eq. (6-13)] with fast neutrons.

* Weinberg, A. M., Fuel Cycles and Reactor Types, *Proc. Intern. Conf. Peaceful Uses Atomic Energy, Geneva*, vol. 3, p. 19, 1955.

REACTOR CLASSIFICATIONS

Reactors can be classified in a number of ways. One classification, based on the intended use, is

1. **Production** of plutonium (see section on converters above)
2. **Power**
3. **Research**, to serve as a source of neutrons and gamma rays
4. **Dual-purpose**, i.e., production and power
5. **Critical experiment**, a very low power assembly intended for determinations of critical masses and flux distributions

Another classification can be made in terms of the fuel.

1. **Natural uranium.** Production reactors are loaded with natural uranium, and as the U-235 is consumed some of the U-238 is converted to Pu-239. Because of competing processes which absorb neutrons, natural uranium can barely sustain a chain reaction, and the permissible arrangements of the fuel and choice of moderators are limited. Heavy water and graphite have been used in natural uranium reactors to date.

2. **Enriched uranium.** Increasing the percentage of U-235 in the mixture allows a much wider choice of moderators, coolants, and structural materials. Some type of facility, such as a gaseous-diffusion plant, must be available to enrich the material. The amount of enrichment varies from a few per cent for large power reactors to greater than 90 per cent for small mobile reactors where the smallest-size core (to reduce size of shielding) is desirable. Research reactors employ highly enriched uranium since it is desirable to provide the maximum neutron flux for a given power dissipation.

3. **Plutonium or Uranium-233.** These would be fueled by the output from breeders after a stock of these materials has been accumulated.

Reactors can be characterized by the neutron energy at which capture in the fuel takes place.

1. **Thermal** reactors are those in which sufficient moderator is present to allow the neutrons to be slowed to thermal equilibrium with the moderator before capture by the fuel. Thermal reactors in general require the smallest amount of fuel.

2. **Intermediate** reactors contain a smaller amount of moderator, and the average neutron energy at which capture takes place is several electron volts above thermal. Operation in this manner eliminates the problem of xenon poisoning and override after shutdown.

3. **Fast** reactors contain no moderator, and fission is caused by fast neutrons. Because fast cross sections for fuel are much smaller than thermal cross sections the critical mass of a fast reactor may be several hundred kilograms vs. 3 to 6 kg for a small thermal reactor. In spite of this and other disadvantages, relative cross sections for competing processes make the fast reactor the most satisfactory design under certain specialized conditions, e.g., breeding from U-238.

A somewhat arbitrary classification can be made in terms of the physical condition of the fuel.

1. A **homogeneous** reactor is one in which the fuel, coolant, and (sometimes) the moderator are intimately mixed.

2. A **heterogeneous reactor** is one in which the fuel is carried in separate containers such as plates, rods, or disks. The fuel may be in the form of a metallic alloy or ceramic pellets (UO_2) contained in metal tubes.

Finally a classification can be made in terms of the coolant employed and its physical condition.

1. **Pressurized water** (not boiling)
2. **Boiling water**
3. **Liquid metals**
4. **Gas**
5. **Fused salts**
6. **Organics**

Several reactors illustrating the classifications outlined above are described in the following pages.

Aqueous Homogeneous Reactor. Figure 6-31 is a simplified schematic diagram of a circulating fuel system.* The fuel is uranyl sulfate (UO_2SO_4) dissolved in H_2O or D_2O . This fuel-moderator-coolant solution is pressurized to prevent boiling ($\sim 1,500$ psi) and is pumped through the reactor core and then the steam generator. The geometry of the system is such that a chain reaction can be sustained only in the core. Steam from the boiler is used to drive a turbine.

A number of practical problems are associated with the operation of this system. About one-third of the radioactive-fission products are gaseous, and these together with a considerable volume of hydrogen and oxygen dissociated by the impact of the fission fragments must continuously be removed from the solution. The hydrogen and oxygen can be recombined, but the radioactive gases must be held up until they have decayed sufficiently to permit release via a stack into the atmosphere. Generally these gases are dried in a cold trap and then absorbed in a charcoal bed. After a year's hold-up the 10-year Krypton-85 will be the principal constituent of this gas residue.

The entire primary loop must be shielded because of the intense radioactivity of the fluid. In addition, because some of the delayed neutrons are carried out of the core with the circulating fluid, the primary piping becomes somewhat activated.

Because heat transfer takes place outside the reactor where adequate transfer surface can be provided independent of reactor-core size limitations, and because xenon is continuously removed, the flux and power density in the core can be extremely high. It is not clear whether hydrodynamic-flow problems or chemical problems associated with the intense radiation are the limiting factors.

* U.S. Atomic Energy Commission, "Reactor Handbook: Engineering," pp. 505 et seq., McGraw-Hill, 1955.

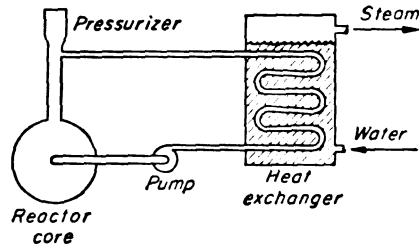


FIG. 6-31. Circulating-fuel aqueous homogeneous reactor. (From "Reactor Handbook: Engineering," Fig. 4.2.26, p. 547, McGraw-Hill, 1955.)

Two-region Breeder or Converter. If the reactor core in the above illustration is surrounded by a blanket of Th-232 the neutrons leaking out of the core will be captured in the thorium and U-233 will ultimately be produced.* The thorium can be in the form of small spheres of ThO₂ or a slurry. The blanket, like the fuel, becomes radioactive, and reprocessing procedures to recover the uranium must be carried out with remotely operated equipment.

Liquid-metal-fuel Reactor. Uranium can be dissolved in liquid bismuth, and the latter substance can be used as a combination coolant and fuel carrier.† Bismuth has the advantage of permitting high temperatures at atmospheric pressure. The reactor is graphite-moderated, and again the construction of the system is such that a chain reaction can be sustained only in the core. A Th-Bi slurry blanket is employed for breeding or conversion. Figure 6-32 is a schematic diagram of the heat-transfer system. The U-Bi solution and the Th-Bi slurry are pumped through heat exchangers where heat is transferred to an intermediate sodium loop. This sodium in turn is circulated through the steam generator. Among other characteristics this system eliminates the problems associated with dissociation of the water in the boiler by gamma rays. The activation of the sodium by the delayed-neutron component in the heat exchanger is small. Fission products and polonium produced by neutron capture in bismuth, rather than activation of the bismuth, determine the level of activity in the primary loop and heat exchanger and the degree of shielding required.

An involved system for continuous reprocessing of the fuel and blanket materials is proposed.‡ This continuous reprocessing substantially reduces the fission-product poisoning effect and the activity after shutdown but does not greatly reduce the primary-loop activity during operation.

Heterogeneous Pressurized-water Reactor. The cooling circuit without the degassing equipment is somewhat similar to the homogeneous-reactor circuit. The fuel is contained in metal-clad plates (stainless steel, zirconium, etc.) and cooled by water circulated under sufficient pressure to inhibit boiling with a reasonable margin to spare.§ This water serves also as the moderator. It is pumped through a steam generator and the steam taken to the usual turbine. In contrast to the circulating-fuel homogeneous reactor the primary heat-transfer surface (fuel plates to water) is inside the reactor core, and the area which can be made available is thus determined by the reactor dimensions. The heat-transfer rate and the power density are set by the allowable film drop. This constraint plus the problem of xenon poisoning generally limits the neutron flux in heterogeneous thermal-power reactors to values below 10^{14} neutrons/cm²-sec.

The primary cooling system is filled with demineralized water. During normal operation fission products are retained in the fuel plates and the most significant

* Briggs, R. B., and J. A. Swartout, Aqueous Homogeneous Reactors, *Proc. Intern. Conf. Peaceful Uses Atomic Energy, Geneva*, vol. 3, p. 175, 1955.

† Raseman, L. J., and J. Weisman, Liquid Metal Fuel Reactor Processing Loops, *Chem. Eng. Progr. Symposium Ser.*, vol. 3, pp. 12, 228, 1954.

‡ Miles, F. T., and C. Williams, Liquid Metal Fuel Reactor, *Proc. Intern. Conf. Peaceful Uses Atomic Energy, Geneva*, vol. 3, p. 125, 1955.

§ Simpson, J. W., et al., Pressurized Water Reactor, *Proc. Intern. Conf. Peaceful Uses Atomic Energy, Geneva*, vol. 3, p. 211, 1955.

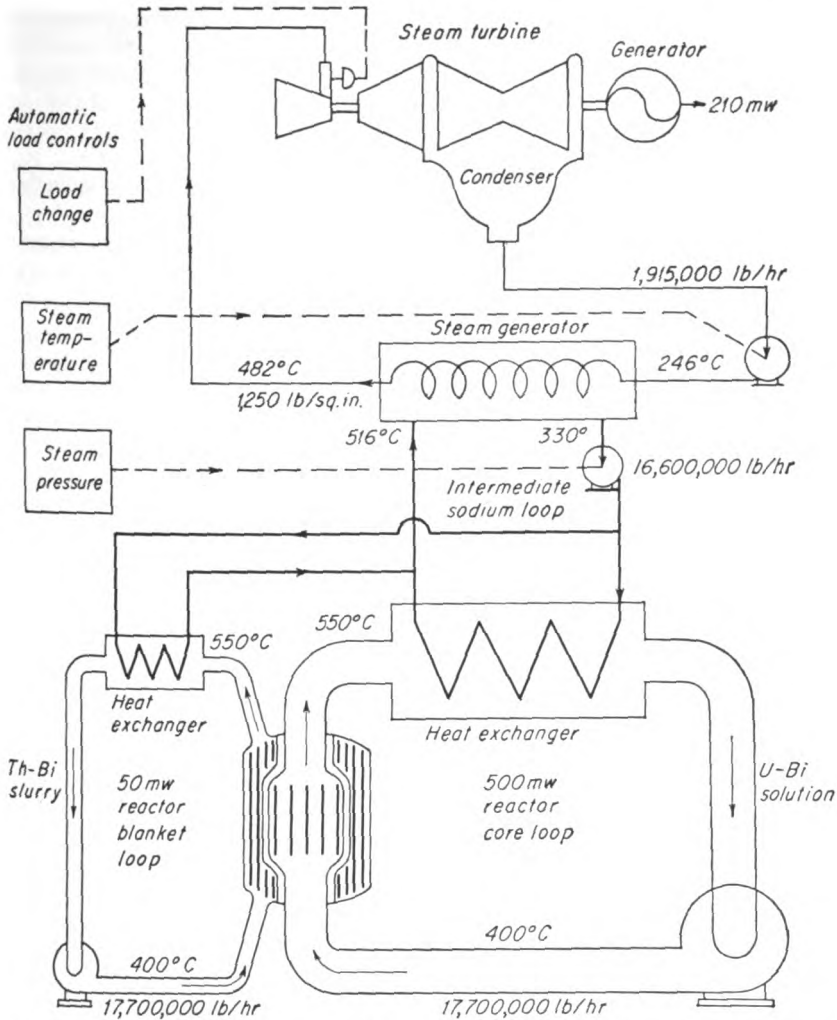


FIG. 6-32. Liquid-metal-fuel reactor. (From *Proc. Intern. Conf. Peaceful Uses Atomic Energy*, Geneva, vol. 3, p. 128, Fig. 3.)

activity is the 7-sec Nitrogen-16 isotope. Decay of the N-16 releases one 6.5-Mev gamma. The activity arises from the $O-16(n,p)N-16$ reaction which has a cross section of 1 barn and a threshold of 9 Mev. An equivalent **effective cross section** per fission neutron is approximately 10^{-5} barn.* The activation of the water at any point in the line leaving the reactor can be computed from

$$A \simeq \frac{\phi \sigma (1 - e^{-\lambda T_1}) e^{-\lambda T_2}}{3 \times 10^{10}} \quad (6-17)$$

* Oak Ridge Natl. Lab. Rept. 963 (declassified), 1952.

where A = activity, curies/cm³

ϕ = fast flux, neutrons/cm²-sec

λ = decay constant of N-16, sec⁻¹

T_1 = exposure time, sec

T_2 = decay time, sec

σ = effective cross section for fast neutrons, barns

In a thermal light-water-moderated reactor ϕ is of the same order of magnitude as the thermal flux, $\lambda = 0.092$ sec⁻¹, and σ is about 10^{-5} barn.

Part of the primary-coolant flow is bypassed through a demineralizer to maintain water purity. Activated corrosion products from the reactor accumulate in the resins and the column must be shielded. Also, remote-handling facilities must be provided if the contaminated resins are to be removed.

Heterogeneous Boiling-water Reactor. In its simplest form the boiling reactor consists of an assembly of heterogeneous fuel elements immersed in a tank of water (Fig. 6-33). The water serves as a moderator as well as the working fluid. The

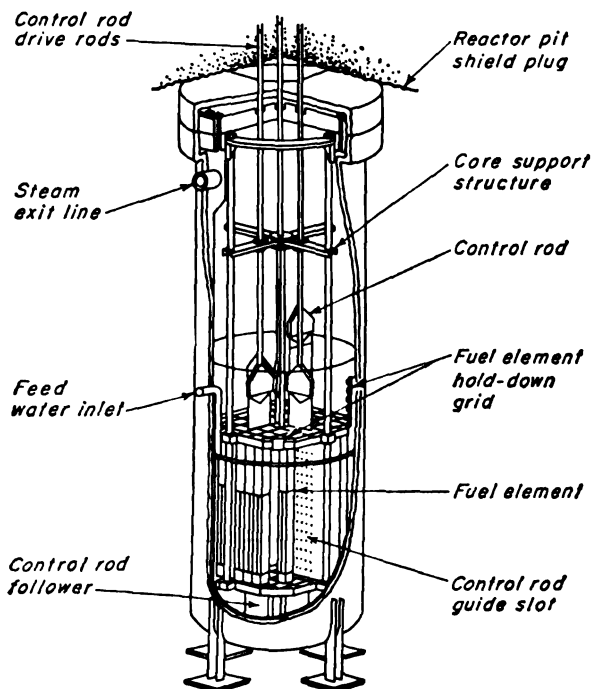


FIG. 6-33. Core assembly in boiling reactor vessel. (From Intern. Conf. Peaceful Uses Atomic Energy—Power Reactors, vol. 3, p. 59, Fig. 3.)

reactor is operated at such a power level that the water boils (about 10 per cent of the core volume is taken up by steam voids) and the steam is taken directly to the turbine. Since the steam contains N-16 activity, as well as a very small amount of solid carry-over, the steam line, the turbine, and the hot well must be shielded.

The comments relative to the water-purification system of the pressurized-water reactor also hold for the boiling-water reactor.

Organic Moderated and Cooled Reactor. Certain organics, notably the terphenyls, appear to be suitable as combined moderators and coolants in a power reactor. These compounds contain approximately the same hydrogen density as water, and hence reactors employing them have nearly the same dimensions as pressurized-water designs but because of the absence of oxygen, the coolant leaving the reactor does not show the N-16 activity [see Eq. (6-17)]. Less shielding is required around the external parts of the primary loop.

Reactor design is similar to the pressurized-water type, but because terphenyls can be selected with low vapor pressures, the primary circuit can be operated at an order of magnitude lower pressure for a given steam temperature. This material has the further advantage of being noncorrosive. However, the phenyls are subject to radiation damage, and provision must be made for removal of decomposition products and addition of fresh material.

Sodium-cooled Graphite Reactor.* An outline drawing of this system is shown in Fig. 6-34. It consists of a graphite-moderated sodium-cooled thermal reactor

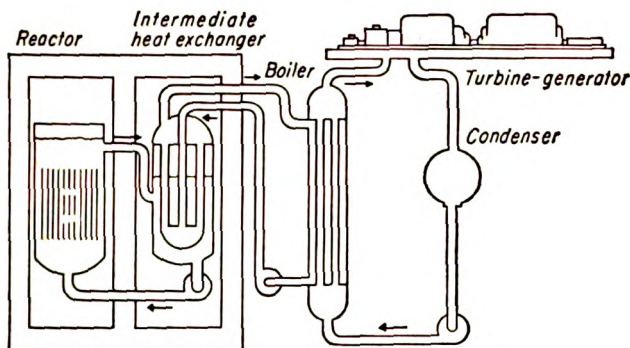


Fig. 6-34. Sodium-cooled graphite reactor. ("Forum Report on Commercial and International Developments in Atomic Energy," p. 43, Fig. 6.)

fuelled with rods of slightly enriched uranium, a sodium-to-sodium heat exchanger, and a sodium-to-water steam generator. Sodium has a neutron-activation cross section of 0.5 barn and a half-life of 15 hr, and hence the primary sodium loop requires extensive shielding. The intermediate sodium loop is not radioactive and serves to isolate the radioactive sodium from the boiler, thus eliminating the problem of dissociation of the water by radiation.

Heterogeneous Fast-breeder Reactor. The economy of the plutonium breeder is best when fissions are caused by neutrons of the highest possible energy. This reactor † employs plutonium or enriched uranium as fuel, and a blanket of natural

* Starr, C., Sodium Graphite Reactor, *Proc. Intern. Conf. Peaceful Uses Atomic Energy, Geneva*, vol. 3, p. 98, 1955.

† Barnes, A. H., et al., EBR II, *Proc. Intern. Conf. Peaceful Uses Atomic Energy, Geneva*, vol. 3, p. 330, 1955.

or depleted uranium surrounds the core. No moderator is present and liquid-metal coolant is used to minimize moderation of the fast neutrons.

Plutonium is extremely toxic, and one of the chief problems associated with this design is development of equipment for reprocessing the plutonium and fabricating it into new fuel elements.

Open-circuit Gas-cooled Heterogeneous Reactor. An example is a graphite-moderated reactor with the fuel in the form of cylindrical cans of natural uranium clad in aluminum. A sketch of the Oak Ridge National Laboratory pile is shown in Fig. 6-35. Air for cooling is drawn through filters to remove dust, then through

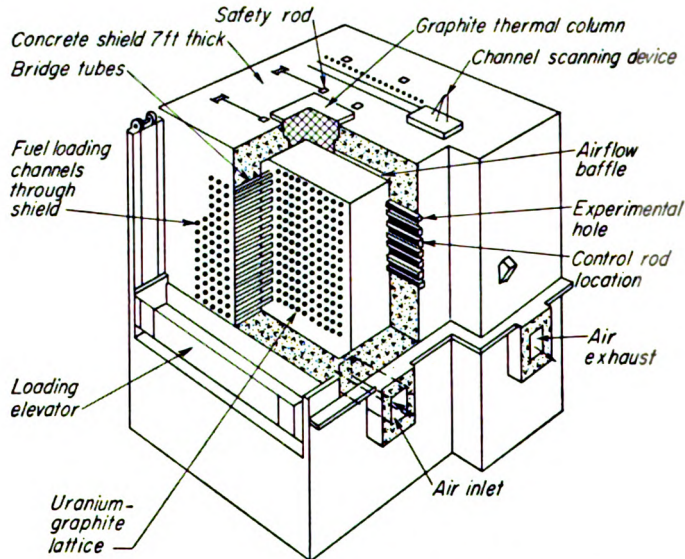


FIG. 6-35. ORNL graphite reactor. (From Glasstone, S., "Principles of Nuclear Engineering," Van Nostrand, 1955.)

horizontal channels in the graphite in which the uranium slugs are placed, then through another set of filters, and finally is discharged through a stack. The Brookhaven reactor,* which is of somewhat similar design, when operating at a power level of 28,000 kw requires an air flow of 270,000 cu ft/min.

Argon comprises about 1 per cent of the atmosphere and is the most abundant radioactive product discharged from an air-cooled reactor. It has a thermal cross section of 0.6 barn and a half-life of 110 min. Because of its rapid decay the area receiving significant radiation is relatively small, and since it is chemically inert assimilation by biological processes is not a problem. In the Brookhaven reactor stack the activity is about 13×10^{-6} curie/cu ft at full power. Stack height must be such that the activity is diluted sufficiently before it reaches the ground to be harmless.

* U.S. Atomic Energy Commission, "Reactor Handbook: Engineering," p. 484, McGraw-Hill, 1955.

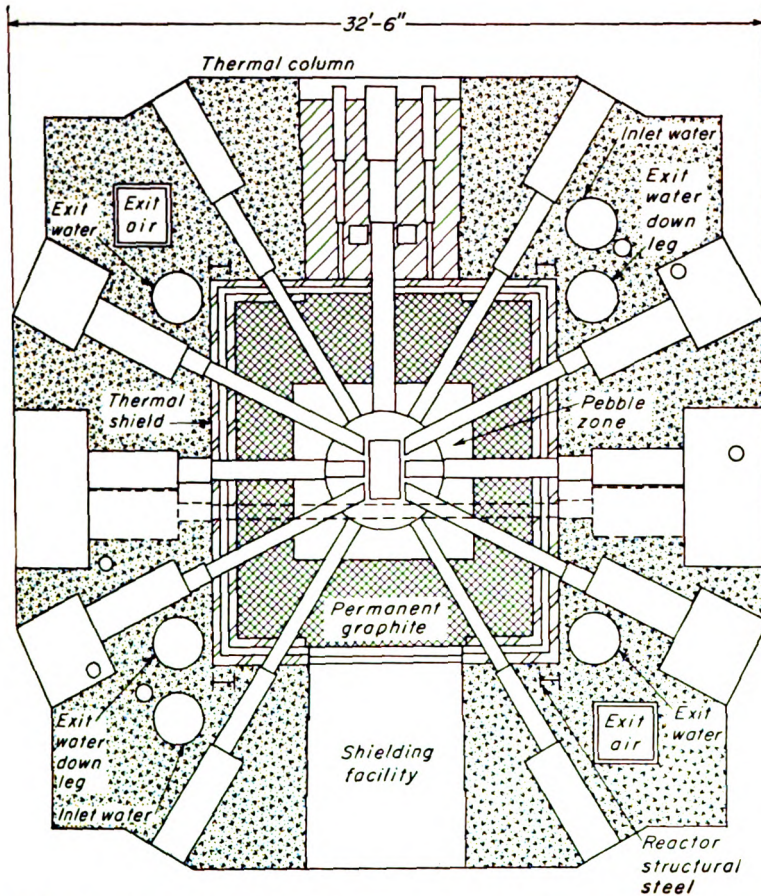


FIG. 6-36. Materials-testing reactor. (From "Research Reactors," p. 158, McGraw-Hill, 1955.)

Closed-circuit Gas-cooled Heterogeneous Reactors. A power-reactor design employs an inert gas (CO_2) and a closed circuit. The CO_2 passes through the reactor and then through a steam generator similar to a standard water-tube boiler. The entire gas circuit must be shielded. In addition to the carbon and nitrogen activities a small quantity of fission-product gases will diffuse through fuel-element cladding if the elements are operated at a high temperature.

Designs have been proposed in which the gases at very high temperatures drive a gas turbine.

Water-cooled Heterogeneous Thermal Research Reactor. The usual objective in research-reactor design is to obtain maximum neutron flux at a given power level. High fluxes imply a small core and very high heat transfer per unit area. The Materials Testing Reactor * employs a core assembled from thin plates of alumi-

* U.S. Atomic Energy Commission, "Research Reactors," pp. 153 *et seq.*, McGraw-Hill, 1955.

num-uranium alloy cooled by rapidly flowing water. By accepting a relatively high film drop the designers have reached an average heat-transfer rate of 300,000 Btu/sq ft-hr. The average thermal flux when the power level is 40,000 kw is 2.5×10^{14} neutrons/cm²-sec.

The core is surrounded by a beryllium and graphite reflector (Fig. 6-36), an iron thermal shield to absorb gamma-ray heating, and a high-density concrete shield. The beam holes penetrate this shield as shown. When not in use the beam holes must be plugged and exhausted into the stack to remove any argon-41 and radioactive dust. Any experiments placed in the holes must also incorporate a plug for neutron and gamma shielding.

The spent fuel elements, which are intensely radioactive, are dropped through a lock in the bottom of the reactor tank into a long pool of water (the "canal"). By this scheme the water shields the fuel elements at all times.

The graphite reflector is cooled by forced circulation of air which is exhausted through a stack. The filtering and dispersal problems are the same as for the gas-cooled reactor (q.v.). Cooling-water shielding designs are similar to those encountered in the pressurized-water reactor.

Open-pool Heterogeneous Research Reactors. In the simplest form this facility consists of an assembly of aluminum-uranium fuel plates suspended in a pool of light water.* Figure 6-37 is a sketch of one arrangement. Convection cooling is

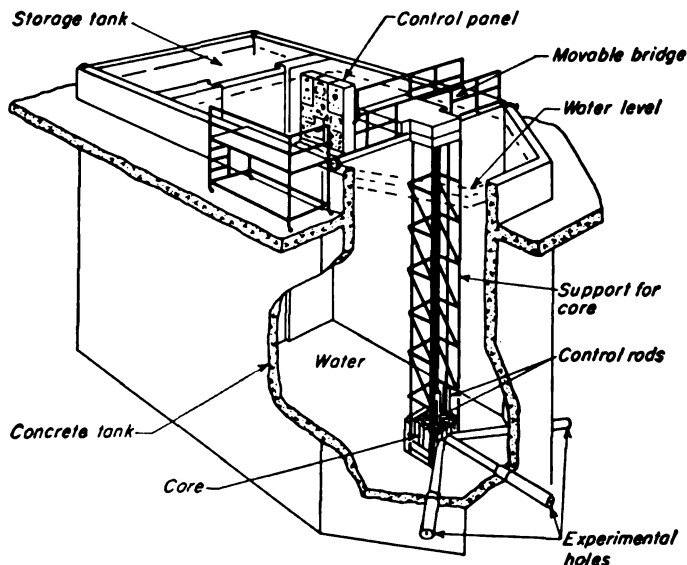


FIG. 6-37. Swimming-pool reactor. (From Glasstone, S., "Principles of Nuclear Reactor Engineering," p. 817, Van Nostrand, 1955.)

satisfactory at power levels of a few hundred kilowatts; above that forced cooling is indicated. Shielding in the horizontal direction is furnished by water plus concrete

* Breazeale, W. M., et al., Swimming Pool Reactor, *Intern. Conf. Peaceful Uses Atomic Energy*, Geneva, vol. 2, p. 420, 1955.

pool walls and in the vertical direction by water. Since neutrons are attenuated much more rapidly in light water than gamma rays the depth is determined by the gamma shielding required. Near the reactor the attenuation length is about 9 in.,* but 20 ft away, because of hardening of the spectrum, this length is about 14 in. Accordingly, a power level of 100 kw requires $16\frac{1}{2}$ ft of water above the top of the core and 1,000 kw about $19\frac{1}{2}$ ft to reduce the surface dose below 7.5 mr/hr. This implies that the N-16 activity is not allowed to rise to the surface before it is substantially decayed.

If the open-pool reactor is operated at power levels greater than 1,000 kw, certain second-order effects assume importance and increase the difficulty of operation. One of these is the formation of radioactive sodium resulting from fast-neutron capture in the aluminum. The reaction is $\text{Al-27}(n,\alpha)\text{Na-24}$. This reaction has an average cross section of 6×10^{-4} barn for fission neutrons and the sodium a half-life of 14 hr.† At 5 megawatts the equilibrium concentration of Na-24 has been calculated to produce a dose rate of 20 to 100 mr at the surface of a 75,000-gal pool. The chief uncertainty in the calculations appears to be in determining the distance which the sodium ion will diffuse through aluminum.

The Water Boiler. This is the original of all enriched reactors.‡ It consists essentially of a stainless-steel sphere (or cylinder) containing an aqueous solution of a uranium salt, generally UO_2SO_4 . The core is enclosed in a graphite reflector and a concrete biological shield (Fig. 6-38). The solution is at atmospheric pressure.

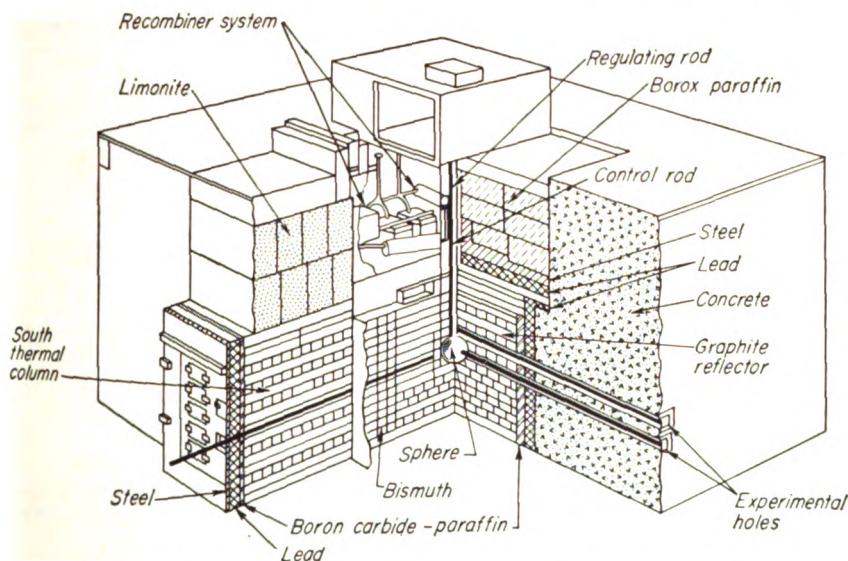


FIG. 6-38. Water-boiler reactor. (From Glasstone, S., "Principles of Nuclear Reactor Engineering," p. 824, Fig. 13.23, Van Nostrand, 1955.)

* Oak Ridge Natl. Lab. Rept. 1891, 1955.

† Hughes, D. J., "Pile Neutron Research," p. 100, Addison-Wesley, 1953.

‡ King, L. D. P., Water Boiler Reactors, *Proc. Intern. Conf. Peaceful Uses Atomic Energy, Geneva*, vol. 2, p. 372, 1955.

Stainless-steel coils through which cooling water flows are immersed in the solution. The cooling water must be held up long enough to allow the N-16 activity to decay before discharge. About $\frac{2}{3}$ cu ft of hydrogen and oxygen are evolved per kilowatt-hour of reactor operation. Details of handling these two gases plus the gaseous-fission products vary with different designers, but all provide for the addition of helium or air to dilute the mixture and to flush it through the recombiner and drier. Sometimes an iodine accumulator is provided ahead of the recombiner. Excess gas pressure is bled through a stack, and monitoring of the stack gases is necessary to ensure against overexposure. Power is limited by the rate of evolution of gas in the solution to levels below 50 kw.

RADIOACTIVE ISOTOPES AND THEIR PROPERTIES

Compiled and Edited by Naomi A. Halden

This compilation is based principally on the work of J. M. Hollander, I. Perlman, and G. T. Seaborg, which appeared in the *Reviews of Modern Physics* (vol. 25, no. 2, pp. 469-651, April, 1953) and with the authors' permission, with additions and corrections based upon more recent works compiled in part by the staff of the Robert A. Taft Sanitary Engineering Center and appearing in the "Radiological Health Handbook" (U.S. Department of Health, Education and Welfare), Simon Kinsman, editor.

Isotope Z A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
				Particles	Gamma transitions
^1_0n		β^-	12.8 m 10-30 m	0.78 p- β^- spect coinc	
^1_1H	99.9849- 99.9861 (diff sources) 99.9851				
^2_1H	0.0139-0.0151 (diff sources) 0.0149				
^3_1H		β^-	12.46 y genet 12.4 y sp act 12.1 y genet	0.01795 spect 0.0183 ion ch 0.0194 spect 0.0189 ion ch 0.0186 0.0180 abs brems- strahlung Zr, Ta	No γ
^3_2He	1.3×10^{-4} (atmos), 1.7×10^{-5} (wells)				
^4_2He	~ 100				
^6_2He		β^-	0.823 s 0.82 s 0.84 s 0.85 s	3.50 spect 3.2 abs 3.5 abs	No γ

Isotope Z	A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
					Particles	Gamma transitions
${}^6_3\text{Li}$		7.52				
Li^{7m}			IT	5.2×10^{-14} s Doppler broadening		~ 0.48 spect
Li^7		92.48				
Li^8			β^- , 2α	0.825 s 0.88 s 0.85 s 0.89 s	β^- : 13 ($\sim 90\%$), ~ 6 ($\sim 5\%$), ~ 3 ($\sim 5\%$) spect < 13 ($\sim 2\%$) β - γ coinc abs 12.0 cl ch, abs two α 's: total energy 3.2, 7-9 cl ch	No γ γ (very weak) β - γ coinc
Li^9			β^- , n	0.168 s 0.170 s 0.19 s		
${}^7_4\text{Be}$			EC	52.93 d 53.61 d 54.3 d 54.5 d		0.479 spect 0.478 spect 0.485 spect; coinc abs 0.474 spect γ (11%) (10-13%) (Be^7 formation yield— γ ratio)
Be^8			2α	$< 2 \times 10^{-14}$ s photodis O^{16} emuls $< 5 \times 10^{-14}$ s photodis O^{16} emuls 10^{-16} – 10^{-17} s calc	Energy of each α in center of mass system: 0.039 spect 0.045 spect 0.051 ion ch 0.043 range emuls	
Be^9		100				
Be^{10}			β^-	2.5×10^6 y sp act + mass spect 2.9×10^6 y yield	0.555 spect 0.560 spect abs 0.553 ion ch	No γ
${}^8_5\text{B}$			β^+ , 2α	0.46 s 0.61 s	β^+ : 13.7 abs spect	
B^{10}		18.45-18.98				
B^{11}		81.02-81.55				
B^{12}			β^-	0.027 s delay coinc 0.022 s delay coinc	13.43 spect 13.3 abs 12 cl ch ~ 9 ($\sim 4\%$) coinc abs	~ 4.5 β - γ coinc abs
${}^{10}_6\text{C}$			β^+	19.1 s	2.2 abs	0.72 ($\sim 100\%$), 1.05 ($\sim 2\%$) scint spect

Isotope Z A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
				Particles	Gamma transitions
${}^6\text{C}^{11}$		β^+	20.4 m 20.5 m 20.0 m	0.99 spect 0.981 spect 0.95 cl ch	No γ , β - γ coinc
C^{12}	98.892 (limestone CO_2)				
C^{13}	1.108 (limestone CO_2)				
C^{14}		β^-	5,568 y weighted avg. and all by sp act + mass spect	0.155 spect 0.156 spect 0.154 spect 0.155 ion ch No conv. spect E (avg) 0.045 calorimetric	No γ
C^{16}		β^-	2.4 s	8.8 abs	5.5 β - γ coinc abs
${}^7\text{N}^{13}$		β^+ , $\beta^+3\alpha$	0.0125 s delay coinc	β^+ : 16.6 abs α : ~ 4 total energy of three α 's	
N^{13}		β^+	9.93 m 10.1 m 10.2 m	1.24 spect 1.25 spect 1.20 spect 1.22 spect	No γ > 0.135, < 0.700
N^{14}	99.635				
N^{15}	0.365				
N^{16}		β^-	7.35 s 7.5 s 7.3 s	~ 10.3 ($\sim 20\%$), ~ 4.3 ($\sim 40\%$), ~ 3.8 ($\sim 40\%$) cl ch 10, 3.5 abs 10 cl ch	γ_1 6.13, γ_2 7.10 (γ_2/γ_1 0.08) pair spect 6.2, 6.7 abs sec, cl ch pair
N^{17}		β^-, n	4.14 s 4.15 s	β^- : 3.7 β -recoil coinc abs n: 0.9 (mean) O^{16} recoil in ion ch 1.0 (mean) p recoil in cl ch	
${}^8\text{O}^{14}$		β^+	76.5 s	1.8 abs	2.3 coinc abs sec
O^{15}		β^+	118.0 s 126 s 127 s	1.683 spect 1.68 abs	No γ
O^{18}	99.759 (air O_2) $\text{O}^{18}/\text{O}^{16}$ variation $\leq \sim 4\%$				

Isotope Z	A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
					Particles	Gamma transitions
^{17}O		0.037 (air O_2)				
^{18}O		0.204 (air O_2)				
^{19}O			β^-	29.4 s 29.5 s 27.0 s	4.5 (30%), 2.9 (70%) abs 4.1 abs ~ 3.2 abs	1.6 abs
^{17}F			β^+	70 s 60 s 66 s 72 s 74 s	1.72 spect 1.7 abs 2.1 cl ch	No γ
^{18}F			β^+	112 m 115 m 107 m	0.649 spect 0.635 spect	No γ
^{19}F		100				
^{20}F			β^-	10.7 s 12 s	5.41, no $\sim 7 \beta^-$ (lim 1%) spect 5.33 (97%), 6.7 (3%) spect 5.0 spect, abs	1.631, no 2.5 γ (lim 0.2%) spect, Be- γ -n reaction 1.64, no 2.5 γ spect, γ - γ coinc 1.64, 2.5 (weak) spect, abs sec 2.2 cl ch recoil γ (coinc with 5.0 β^-) β - γ coinc
^{19}Ne			β^+	18.2 s 18.5 s 20.3 s	2.18 spect 2.2 cl ch 2.3 abs	No γ
^{20}Ne		90.92				
^{21}Ne		0.257				
^{22}Ne		8.82				
^{23}Ne			β^-	40.2 s 40.7 s 40 s	4.21 ($\sim 93\%$), 1.18 ($\sim 7\%$) spect 4.3 abs 4.1 abs	~ 3 abs
^{20}Na			β^+, α	0.385 s 0.23 s	$3.5 < \beta^+ < 7.3$ est	
^{21}Na			β^+	22.8 s 23 s	2.50 spect 2.53 spect	No γ
^{22}Na			$\beta^+ \sim 100\%$, no EC	2.60 y 2.8 y	0.542 spect 0.575 spect ~ 1.8 (0.06%) spect ~ 1.8 (0.004%) cl ch	1.277 spect 1.30 (coinc with β^+) spect, β - γ coinc 1.3 spect
^{23}Na		100				

Isotope Z	A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
					Particles	Gamma transitions
^{24}Na			β^-	15.06 h	1.390 spect, coinc	γ_1 1.3679, γ_2 2.7535
				15.04 h	1.4 spect	spect
				15.10 h	4.17 (0.003%),	γ_1 1.380, γ_2 2.758
				15.0 h	no 5.5 β^- , spect	spect
					lim 4.15 β^- , 0.01% spect	1.380, 2.765 spect γ_2 (s/γ 3×10^{-6}) 2.748 spect 2.755 spect γ_1 (coinc with γ_2) spect, β - γ , γ - γ coinc; γ - γ coinc abs 3.7 (0.04%) D- γ -p ion ch ~ 4 (0.05%) spect
^{25}Na			β^-	58 s	3.7 ($\sim 55\%$),	>0.5 (weak) abs
				62 s	2.7 ($\sim 45\%$) abs	
				60 s	3.4 abs	
					3.3 abs	
^{23}Mg			β^+	11.9 s	2.99 scint spect	No γ
				12.3 s	2.8 cl ch	
				11.6 s		
^{24}Mg	78.60					
^{25}Mg	10.11					
^{26}Mg	11.29					
^{27}Mg			β^-	9.45 m	1.80 (80%),	γ_1 1.01, γ_2 0.84 (γ_1 coinc with γ_2) spect, γ - γ coinc 1.02, 0.84, 0.64 spect 1.05 cl ch recoil
				9.6 m	0.9 (20%) spect	
				10.0 m	1.8 cl ch	
				10.2 m	1.7 abs	
					~ 1.8 (coinc with γ) β - γ coinc	
^{28}Mg			β^-	21.2 h	0.3-0.4 abs	<0.1 abs
^{24}Al			β^+ or EC, $\alpha(p?)$	2.3 s		
^{26}Al			β^+	7.3 s		
^{28}Al			β^+	6.5 s	2.8 abs	
				6.3 s	3.0 cl ch	
				7.0 s	3.4 abs	
				7.2 s		
^{27}Al	100					
^{28}Al			β^-	2.27 m	2.865 spect	1.782 spect
				2.07 m	2.75 (coinc with γ)	1.80 spect
				2.30 m	coinc abs	1.80 abs sec
					3.01 spect No 4.6 β^-	1.82 spect
^{29}Al			β^-	6.56 m	2.5 ($\sim 70\%$), 1.4	1.2 ($\sim 80\%$) (coinc with 2.5 β^-), 2.3
				6.7 m	($\sim 30\%$) (both coinc with γ) β - γ coinc abs ~ 2.5 cl ch, abs	($\sim 20\%$) coinc abs sec

Isotope Z	A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
					Particles	Gamma transitions
$^{14}\text{Si}^{27}$			β^+	4.9 s 4.5 s 5.4 s	3.48 scint spect 3.5 cl ch 3.7 cl ch	
Si^{28}	92.27					
Si^{29}	4.68					
Si^{30}	3.05					
Si^{31}			β^-	2.62 h 2.65 h 2.6 h 2.8 h 2.7 h	1.471 spect 1.486 spect 1.48 abs	0.17, 0.52, ~ 1 (weak) (?) abs No γ
$^{13}\text{P}^{29}$			β^+	4.57 s 4.6 s	3.6 cl ch	1.28 (2.5%), 2.42 (0.5%), scint spect, γ - γ coinc
P^{30}			β^+	2.55 m 2.18 m	3.5 spect 3.4 abs 3.0 cl ch	
P^{31}	100					
P^{32}			β^-	14.30 d 14.35 d 14.07 d 14.60 d	1.701 spect E (avg) 0.70 ion ch	No γ
P^{33}			β^-	25.4 d 24.8 d 25 d	0.27 spect 0.26 spect 0.25 abs	No γ
P^{34}			β^-	12.4 s 12.7 s	5.1 (75%), 3.2 (25%) abs	
$^{16}\text{S}^{31}$			β^+	3.18 s 3.2 s 2.6 s	3.85 cl ch 3.87 cl ch 4.1 scint spect	
S^{32}	95.018 (meteoritic Sulfur) terrestrial $\text{S}^{32}/\text{S}^{34}$ variation $\leq 5\%$					
S^{33}	0.750 (meteoritic Sulfur)					
S^{34}	4.215 (meteoritic Sulfur)					
S^{35}			β^-	87.1 d 88 d	0.1670 spect 0.167 spect abs 0.169 spect abs 0.168 spect 0.166 spect 0.168 ion ch	

Isotope Z A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
				Particles	Gamma transitions
$^{36}_{16}\text{S}$	0.017 (meteoritic Sulfur)				
S^{37}		β^-	5.04 m 5.0 m	4.3 (10%), 1.6 (90%) abs 1.4, ~ 4 abs	2.7 abs sec 2.75 abs sec
$^{33}_{17}\text{Cl}$		β^+	2.8 s 2.4 s 1.8 s	4.13 cl ch 4.4 scint spect	
Cl^{34}		β^+	33.2 m 33.0 m 33 m	4.5 (46%), 2.6 (28%), 1.3 (26%) spect 5.1 ($\sim 80\%$), 2.4 ($\sim 20\%$) cl ch	3.22, 2.10, 1.16 scint spect 3.3, 2.1, 0.145 spect, spect conv 3.4 cl ch recoil
Cl^{35}	75.4				
Cl^{36}		β^- No β^+ (lim 0.01%) (lim 0.03%)	4.4×10^5 y sp act 3.6×10^5 y sp act 2.0×10^5 y yield 2×10^6 y yield $\sim 10^6$ y yield	0.714 spect 0.66 abs 0.64 abs	No γ
Cl^{37}	24.6				
Cl^{38}		β^-	37.29 m 37.5 m 37.0 m 38.5 m	4.81 (53%), 2.77 (16%), 1.11 (31%) spect 5.0, 2.8, 1.1 spect; spect, coinc abs 5.2 (53%), 2.70 (11%), 1.19 (36%) spect	γ_1 2.15, γ_2 1.60 ($\gamma_1/\gamma_2 \approx 1.3$) spect γ_1 2.19, γ_2 1.64 (γ_1 coinc with γ_2) spect, coinc 2.15, 1.65 spect No 3.75 γ (lim 0.03%) Be- γ -n reaction
Cl^{39}		β^-	55.5 m ~ 1 h	1.65 (93%), 2.96 (7%) abs	1.35, 0.35 (coinc with 1.65 β^-) coinc abs sec
$^{35}_{18}\text{A}$		β^+	1.88 s 1.84 s	4.38 cl ch 4.41 cl ch	
A^{36}	0.337				
A^{37}		EC EC (L/K 0.087)	35.0 d 34.1 d		No γ , Cl $\text{K}_\alpha - x$
A^{38}	0.063				
A^{39}		β^-	~ 265 y sp act	0.565 spect	No $\gamma > 0.3$ (lim 0.1%)
A^{40}	99.600				

Isotope Z	A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev				
					Particles	Gamma transitions			
$^{18}\text{Ar}^{41}$			β^-	109 m	1.245 ($\sim 100\%$) spect	1.37 cl ch recoil			
				110 m	1.18 (99.3%), 2.5 (0.7%) abs, coin abs	1.3 (coinc with 1.2 β^-) β - γ coin, abs sec γ ($e/\gamma \sim 0$) spect conv			
Ar^{42}			β^-	≥ 3.5 y					
$^{19}\text{K}^{37}$			β^+	1.3 s 1.2 s	4.6 scint spect				
K^{38}			β^+	7.7 m	2.8 spect	2.16 scint spect			
				7.5 m 7.6 m	2.5 abs	~ 2.1 abs sec			
K^{39}	93.08 $\text{K}^{39}/\text{K}^{41}$								
K^{40}	0.0119	β^- 89%, EC 11%	t_{β^-} sp act (uncorr. for EC): 1.32×10^9 y 1.26×10^9 y 1.29×10^9 y 1.49×10^9 y 1.42×10^9 y $t_{1/2} 1.2 \times 10^9$ y calc from avg t_{β^-} and EC/β^-	1.33 spect	1.46 scint spect				
		$\beta^- \sim 88\%$, EC (K) $\sim 12\%$		1.36 spect	1.48 scint spect				
		$\beta^- \sim 93\%$, EC (K) $\sim 7\%$		1.36 scint spect	1.47 scint spect				
		assuming EC/ $\gamma = 1$,		1.35 spect coin	1.54 (with EC) abs, coin				
		β^- 90%, EC 10%		1.41 abs	1.55 abs				
		β^- 91%, EC 9%		1.28 scint spect	γ (with EC)				
		β^- 89%, EC 11%			γ/β^- 0.12 sp act EC/ $\gamma \sim 1$				
		β^- 94%, EC 6%							
		No β^+ (lim 0.06%)							
		No β^+ (lim 0.002%)							
		$\beta^+ \leq 0.1\%$							
		K^{41m}			IT		6.7×10^{-9} s delay coin		~ 1.3 scint spect
		K^{41}		6.91					
K^{42}		β^-	12.44 h 12.5 h 12.4 h	3.58 (75%), 2.04 (25%) spect	1.51 spect				
				3.60, 1.9 spect	1.50 spect				
				3.5 ($\sim 70\%$) (not coin with γ), ~ 1.8 ($\sim 30\%$) abs, coin	1.5 (17%) scint spect				
K^{43}		β^-	22.4 h	0.81, 0.24 spect, abs	~ 0.4 abs				
K^{44}		β^-	18 m						
$^{20}\text{Ca}^{49}$		β^+	1.06 s	5.1 scint spect					
			1.1 s						
Ca^{40}	96.97								
Ca^{41}		EC	1.1×10^5 y yield		$K K_{\alpha} - x$				

Isotope Z	A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
					Particles	Gamma transitions
⁴² Ca		0.64				
Ca ⁴³		0.145				
Ca ⁴⁴		2.06				
Ca ⁴⁶			β^-	152 d 180 d	0.254 spect 0.255 scint spect 0.260 abs 0.26 abs E (avg) 0.075 ion ch	No γ
Ca ⁴⁶		0.0033				
Ca ⁴⁷			β^-	4.8 d 5.8 d	1.2 abs 1.1	1.3 abs
Ca ⁴⁸		0.185		$t_{\beta^-} > 2 \times 10^{16}$ y sp act		
Ca ⁴⁹			β^-	8.5 m	~2.7 abs	Hard
⁴¹ Sc			β^+	0.873 s 0.87 s	4.94 cl ch	
Sc ⁴³			β^+	3.92 h 4 h	1.18 (72%), 0.77 (28%) spect 1.12 abs, spect 1.4 cl ch, abs	0.375, no higher γ (lim 15%) spect 1.65 abs 1.0 abs
Sc ^{44m}			IT	2.44 d 2.4 d 2.2 d		0.271 spect, spect conv 0.269 spect conv 0.26 spect conv 0.28 abs conv
Sc ⁴⁴			β^+ , EC	3.92 h 4.0 h 4.1 h	1.463 spect 1.45 spect 1.5 abs	1.16 spect, spect conv 1.18 coinc abs sec
Sc ⁴⁵		100				
Sc ^{46m}			IT	19.5 s		0.142 (K/L ~10) spect conv 0.135 scint spect
Sc ⁴⁶			β^- , no EC β^- , EC (weak) No β^+ (lim 0.0016%)	85 d 84 d	0.36 spect 0.36 abs 0.34 abs 1.2 (~0.5%) spect No 1.5 β^- (lim 0.05%) cl ch No 1.5 β^- (lim 0.06%) spect 1.5 (~2%) spect, abs	γ_1 0.89, γ_2 1.12 spect, spect conv 0.88, 1.12 spect 0.90, 1.12 spect γ_1 (e/γ 1.7×10^{-4}), γ_2 (e/γ 0.98 $\times 10^{-4}$) spect conv γ_1 (e/γ 1.4×10^{-4}), γ_2 (e/γ 0.61 $\times 10^{-4}$) spect conv γ_1 (coinc with 0.36 β^- and γ_2) β - γ , γ - γ coinc γ_1 (coinc with 1.5 β^- and γ_2) delay coinc

Isotope Z A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
				Particles	Gamma transitions
$^{47}_{21}\text{Sc}$		β^-	3.43 d	0.61 abs	γ
$^{48}_{21}\text{Sc}$		β^-	44 h	0.64 spect 0.57 abs	γ_1 1.33, γ_2 0.98, γ_3 ~1.0 scint spect, γ - γ coin 1.32, 0.99, no 2.3 γ (lim 0.1%) scint spect 1.33, 0.98 spect 1.35 spect 1.33 abs
$^{49}_{21}\text{Sc}$		β^-	57 m	1.8 abs	No γ
$^{43}_{22}\text{Ti}$			0.6 s		
$^{45}_{22}\text{Ti}$		β^+ , EC	3.09 h 3.05 h 3.08 h	1.02 ($\geq 96\%$), 0.57 ($\leq 4\%$) spect 1.00 spect 1.2 cl ch	0.45 (weak), (no 0.8 γ) spect 0.80 spect, abs
$^{46}_{22}\text{Ti}$	7.95				
$^{47}_{22}\text{Ti}$	7.75				
$^{48}_{22}\text{Ti}$	73.45				
$^{49}_{22}\text{Ti}$	5.51				
$^{50}_{22}\text{Ti}$	5.34				
$^{51}_{22}\text{Ti}$		β^-	5.82 m 5.75 m 6 m	1.7, 1.35 (coin with 0.32 γ) scint spect 2.2 ($\sim 30\%$), 1.9 ($\sim 70\%$) abs, spect 1.6 abs	0.320, 0.910 (weak) scint spect 0.32 scint spect
$^{46}_{23}\text{V}$		β^+	0.40 s	>6.0 scint spect	
$^{47}_{23}\text{V}$		β^+	33 m	1.7 abs 2.0 abs ~1.9 abs	γ
$^{48}_{23}\text{V}$		β^+ 58%, EC 42% β^+ , EC	16.0 d	0.69 ($\sim 95\%$), ~0.8 ($\sim 5\%$) spect 0.72 spect 0.6 abs 1.0 cl ch	γ_1 0.99, γ_2 1.32 (γ_2 coin with γ_1), γ_3 2.29 (γ_3/γ_2 ≈ 0.017) scint spect γ_1 0.99, γ_2 1.32, γ_3 2.22 scint spect, β^+ - γ , γ - γ coin γ_1 0.98, γ_2 1.33 (γ_2 coin with γ_1 , both coin with β^+) spect, β - γ coin γ_1 (e/γ 2×10^{-4}), γ_2 (e/γ 8×10^{-5}) spect conv γ_3 2.22 ($\sim 2\%$), no 2.3 γ (lim 0.5%) scint spect γ_1 0.99, γ_2 1.32 spect conv γ_1 (coin with γ_2) γ - γ coin

Isotope Z A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
				Particles	Gamma transitions
^{50}V		<i>EC</i>	635 d ~600 d		0.119, 0.081 spect conv
V^{50}	0.24				
V^{51}	99.76				
$\text{V}^{52(\text{m}^?)}$		β^- <i>IT</i>	3.76 m 3.74 m 3.75 m	2.1 abs 2.6 cl ch conv: 0.25, β^- with V^{52} , abs, β^- , β -conv coin	1.5 abs 1.3 abs ~1.5 γ with V^{52}
V^{52}		β^-	2.6 m	2.7 cl ch	1.5 abs
$\text{V}^{52, 50\text{m}}$			16 h		
V^{53}		β^-	23 h	~0.6 abs	
^{54}Cr		<i>EC</i>	~23 h		
Cr^{49}		β^+	41.9 m 45 m	1.45 abs, cl ch	0.18, 1.55 abs
Cr^{50}	4.31				
Cr^{51}		<i>EC</i> No β^+	27.8 d 26.5 d		0.330 (~3%, eK/γ ~0.02), spect, spect conv 0.32 (8%, e/γ very small) scint spect, x - γ coin 0.323 spect, spect conv 0.320 (single γ) spect 0.32 spect conv
Cr^{52}	83.76				
Cr^{53}	9.55				
Cr^{54}	2.38				
Cr^{55}			~2 h		
$^{55}\text{Mn}^{49, 50}$		β^+	0.28 s 0.5 s	>6.3 scint spect	
Mn^{51}		β^+	44.3 m 46 m	2.4 abs 2.0 abs	
$\text{Mn}^{52\text{m}}$		β^+ 99 + %, <i>IT</i> (?) ~0.05%	21.3 m 21 m	2.66 spect 2.2 cl ch	1.46 (coin with β^+) spect, β - γ coin 0.392 (0.05%) (with <i>IT</i> ?) spect conv
Mn^{52}		<i>EC</i> 65%, β^+ 35%	6.0 d 6.2 d 6.5 d 6 d	0.58 spect 0.75 abs 0.77 cl ch	γ_1 0.73, γ_2 0.94, γ_3 1.46 (γ_1 coin with γ_2 and γ_3 , all coin with β^+) spect, β - γ coin
$\text{Mn}^{54\text{m}}$		β^-	2.1 m		
Mn^{54}		<i>EC</i> No β^+ , no β^-	310 d		0.84 spect, x - γ coin 0.85 abs

Isotope Z A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
				Particles	Gamma transitions
²⁵ Mn ⁵⁵	100				
Mn ⁵⁶		β^-	2.576 h 2.59 h	2.81 (50%), 1.04 (30%), 0.65 (20%) spect 2.86 (60%), 1.05 (25%), 0.75 (15%) spect 2.88, 1.04 spect	γ_1 0.822, γ_2 1.77, γ_3 2.06 ($\gamma_1/\gamma_2/\gamma_3$ $\approx 10/3/2$) spect ($\gamma_2/\gamma_3 = 1.0$) spect γ_1 0.85, γ_2 1.81, γ_3 2.13 ($\gamma_1/\gamma_2/\gamma_3$ $\approx 10/3/2$) spect, β - γ coinc ~ 2.7 ($\sim 0.1\%$), ~ 3.0 ($\sim 0.2\%$) D- γ -p ion ch
Mn ⁵⁷		β^-	7 d	1.0 spect	
²⁶ Fe ⁵²		EC 60%, β^+ 40%	8.3 h 7.8 h	~ 0.55 abs ~ 0.6 abs	No γ >0.5 scint spect
Fe ⁵³		β^+	8.9 m	2.5 scint spect 2.6 abs	No γ
Fe ⁵⁴	5.84				
Fe ⁵⁵		EC, no β^+	2.94 y 3.0 y		No γ ~ 0.07 (0.002%), Mn K- α
Fe ⁵⁶	91.68				
Fe ^{57m}		IT	1.1×10^{-7} s delay coinc		0.014 spect
Fe ⁵⁷	2.17				
Fe ⁵⁸	0.31				
Fe ⁵⁹		β^-	45.1 d 47.1 d 45.5 d 46 d	0.460 ($\sim 50\%$), 0.257 ($\sim 50\%$) spect, β - γ coinc abs 0.45, no 0.26 β^- spect	γ_1 1.295, γ_2 1.097 spect γ_1 1.29, γ_2 1.10 (not coinc with γ_1) spect, γ - γ coinc 1.30, 1.10 spect γ_1 (e_K/γ 0.84 $\times 10^{-4}$), γ_2 (e_K/γ 1.45×10^{-4}) spect conv 0.195 (2.5%, coinc with 1.1 γ) scint spect, γ - γ coinc Others
Fe ⁶⁰		β^-	8.4 h	~ 1.5 abs	
²⁷ Co ⁵⁴		β^+	0.18 s	>7.4 scint spect	
Co ⁵⁶		$\beta^+ \sim 60\%$, EC $\sim 40\%$	18.2 h	1.50 ($\sim 50\%$), 1.01 ($\sim 50\%$) spect 1.50 spect	γ_1 0.477 (γ/β^+ 0.3, e/γ 0.0007) γ_2 0.935 (γ/β^+ 1.4, e/γ 0.0005) γ_3 1.41 (γ/β^+ 0.3, e/γ 0.00004) spect, spect conv, β - γ coinc

Isotope Z A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
				Particles	Gamma transitions
^{56}Co		EC, β^+	72 d 80 d	1.50 (coinc with γ_1 and γ_2) spect 1.2 abs	γ_1 0.845, γ_2 1.26 (coinc with γ_1), γ_3 1.74, γ_4 2.01, γ_5 2.55, γ_6 3.25 ($\gamma_1/\gamma_2/\gamma_3/\gamma_4/\gamma_5$ / $\gamma_6 = 1.0/0.5$ / 0.2/0.1/0.2/0.2) spect, β - γ coinc
^{57}Co		β^+	270 d	0.26 abs	0.119, 0.131 spect 0.117 (e/γ large, K/L 7), 0.130 (e/γ large, K/L 7) spect, spect conv With $\text{Fe}^{57\text{m}}$: 0.014 spect
$^{58\text{m}}\text{Co}$		IT, no β^+	9.2 h 9.0 h 8.8 h		0.025 (e/γ large, K/L 1.9) spect conv
^{58}Co		EC 85%, β^+ 15%	72 d	0.47 spect 0.4 abs	0.81 spect, β - γ coinc, 0.81 (e/γ 0.0003) spect conv 0.6 abs
^{59}Co	100				
$^{60\text{m}}\text{Co}$		IT 99+%, β^- 0.28%	10.1 m 10.7 m	1.56 spect 1.35 spect 1.28 spect	0.0589 (K/L 4.6) spect conv 0.059 scint spect 0.056 spect conv
^{60}Co		β^-	5.27 y 5.3 y	0.306 spect 0.318 spect 0.310 spect	γ_1 1.3316, γ_2 1.1715 cryst spect 1.3316, 1.1715 spect γ - γ coinc γ_1 (e/γ 1.24×10^{-4}), γ_2 (e/γ 1.72 $\times 10^{-4}$) spect conv γ_1 (e/γ 1.29×10^{-4}), γ_2 (e/γ $1.73 \times$ 10^{-4}) spect conv γ_1 (e/γ 1.8×10^{-4}), γ_2 (e/γ $2.3 \times$ 10^{-4}) spect conv
^{61}Co		β^-	99.0 m 105 m 110 m	1.42 (~55%), 1.00 (~45%) abs 1.3 abs	~0.5 abs No γ
^{62}Co		β^-	13.9 m	2.3 abs	1.3 (coinc with β^-) abs, β - γ coinc
^{63}Co		β^-	1.6 m		γ
^{64}Co			~5 m		

Isotope Z A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
				Particles	Gamma transitions
$^{56}_{28}\text{Ni}$		EC ~100%, no β^+ (lim 1%)	6.4 d 6.0 d		γ_1 0.17, γ_2 0.28, γ_3 0.48, γ_4 0.81, γ_5 0.96, γ_6 1.33, γ_7 1.58, γ_8 1.75 ($\gamma_1/$ $\gamma_2/\gamma_3/\gamma_4/\gamma_5/\gamma_6/$ $\gamma_7/\gamma_8 \approx 1.0/0.3/$ 0.4/0.8/0.1/0.05/ 0.15/0.02) scint spect 0.14, 0.77, >1.4 scint spect
Ni^{57}		β^+ 50%, EC 50%	36 h	0.835 spect 0.845 spect	γ_1 1.91, γ_2 1.38 ($\gamma_2/\gamma_1 = 6$) spect γ_1 1.90, γ_2 1.39 ($\gamma_2/\gamma_1 = 5$) scint spect 0.128 spect, spect conv 0.120, 1.9 scint spect, spect conv, β - γ coinc
Ni^{58}	67.76				
Ni^{59}		EC	8×10^4 y yield 8×10^5 y yield $\sim 3 \times 10^5$ y yield		Co-K-x, no γ Co-K-x
Ni^{60}	26.16				
Ni^{61}	1.25				
Ni^{62}	3.66				
Ni^{63}		β^-	85 y yield 61 y yield	0.067 ion ch 0.063 ion ch	
Ni^{64}	1.16				
Ni^{65}		β^-	2.564 h 2.6 h	2.10 (57%), 1.01 (14%), 0.60 (29%) spect	1.49, 1.12, 0.37 spect, β - γ , γ - γ coinc
Ni^{66}		β^-	56 h		
$^{58}_{29}\text{Cu}$		β^+	7.9 m 10 m		
Cu^{58}		β^+	3.04 s 2.6 s	>7.5 scint spect	
Cu^{59}		β^+	81 s		
Cu^{60}		β^+	24.6 m	1.8, 3.3 (<5%) abs	1.5 abs
Cu^{61}		β^+ 68%, EC 32% β^+ 72%, EC 28% β^+ 75%, EC 25%	3.33 h 3.4 h 3.3 h	1.205 (96%), 0.55 (4%) spect 1.23 spect	γ_1 0.655 (γ/β^+ 0.25, $e/\gamma \sim 0$), γ_2 0.284 (γ/β^+ 0.05, e/γ 0.015), γ_3 0.076 (γ/β^+ 0.01, e/γ large) spect, spect conv

Isotope Z	A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
					Particles	Gamma transitions
⁶¹ Cu						γ_1 (17%), γ_2 (3%), γ_3 (~0.6%) 0.652, 0.279, 0.070 (K/L ~10) spect, spect conv
⁶³ Cu			β^+	10.1 m	2.92 spect	0.56 abs
				10.0 m	2.83 spect	
				9.9 m		
				10.5 m		
⁶³ Cu	69.1					
⁶⁴ Cu			EC 42%, β^- 39%, β^+ 19% (HPS, calc from avg of K/ β^+ and β^-/β^+) K/ β^+ 2.2 K/ β^+ 2.3 K/ β^+ 1.8 K/ β^+ 2.7 β^-/β^+ 2.0 β^-/β^+ 2.1 $\beta^+ + EC$ β^- 1.62	12.80 h	β^- :	γ (with EC) coin
				12.74 h	0.571 spect	γ (1% of EC)
				12.88 h	0.570 spect	1.34 (weak) spect
				12.8 h	0.58 spect	1.35 (γ/β^+ 0.025) spect
					β^+ :	1.33 scint spect
					0.657 spect	(γ/β^+ 0.023) abs
					0.644 spect	1.38 (γ/β^+ 0.032) abs
					0.65 spect	~1.34 (eK/γ ~1.3 $\times 10^{-4}$) spect
					0.66 spect	conv
						γ (e/γ <0.005) spect
						conv
⁶⁵ Cu	30.9					
⁶⁶ Cu			β^-	5.10 m	2.63 (91%), 1.5 (9%)	1.044 (γ/β^- 0.10, e/γ <3 $\times 10^{-4}$)
				5.12 m	spect	spect, spect conv
				5.17 m	2.7 (~94%), 1.65	1.05 scint spect
				5.18 m	(~6%) scint spect,	
				5.2 m	β - γ coin	
⁶⁷ Cu			β^-	58.5 h	0.38 (67%), 0.57 (33%)	0.191 (e/γ 0.01, K/L ~8), 0.096 (e/γ 0.09, K/L 7.4)
				61.0 h	spect	scint spect, spec conv, β - γ coin
				56 h	0.54 abs	
				61 h		
⁶² Zn			EC ~90%, β^+ ~10%	9.33 h 9.5 h	0.66 spect	0.0418 (K/L >6) spect conv
⁶³ Zn			β^+ 93%, EC 7%	38.3 m	2.36 (92%), 1.40 (7%),	0.960 (~8%, e/γ 2 $\times 10^{-4}$), 1.89 (~4%, e/γ ~0), 2.60 (~0.5%, e/γ ~0) spect, spect conv, abs, γ - γ coin
				38.5 m	0.5 (~1%) spect	
					2.32 spect	
⁶⁴ Zn	48.89					
⁶⁵ Zn			EC (K) 97.5%, β^+ 2.5%, EC 99 + %, β^+ 0.7%	250 d	0.325 spect	1.120 spect conv
				245 d	0.32 spect	1.125 spect, spect conv
						1.114 spect
						1.112 (e/γ 2.3 $\times$ 10 ⁻⁴) spect conv

Isotope Z A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
				Particles	Gamma transitions
³⁰ Zn ⁶⁵ (Cont.)					1.127 scint spect 1.12 spect γ (46% of EC) x - e coin γ (44% of EC) x - γ coin $\gamma/\beta^+ \approx 65$ γ - γ coin 0.21 (coin with β^+ , $e/\gamma \sim 0.1$) β - γ , β -conv coin
Zn ⁶⁶	27.81				
Zn ^{67m}		IT	8.5×10^{-6} s delay coin 9×10^{-6} s delay coin		0.092 (e_K/γ 0.63, K/L 7) scint spect 0.092 (e/γ 0.6) scint spect
Zn ⁶⁷	4.11				
Zn ⁶⁸	18.56				
Zn ^{69m}		IT	13.8 h		0.437 (e/γ 0.049) spect conv 0.436 ($e/\gamma \sim 0.065$) spect conv, scint spect 0.439 ($e/\gamma \sim 0.04$) spect, spect conv 0.440 spect 0.450 ($e/\gamma \sim 0.06$)
Zn ⁶⁹		β^-	57 m 51 m 52 m	0.897 spect 0.914 spect	No γ
Zn ⁷⁰	0.62				
Zn ⁷¹		β^-	2.2 m	2.1 abs	
Zn ⁷²		β^-	49.0 h	~ 0.3 (95%), ~ 1.6 (5%) abs	γ
³¹ Ga ⁶⁵		EC $\beta^+ > 50\%$	15 m	2.2	0.054, 0.117 spect conv
Ga ⁶⁶		β^+ 64%, EC 36%, β^+ 66%, EC 34%	9.45 h 9.4 h 9.2 h	4.144 (87%), 1.4 (4%), 0.88 (7%), 0.40 (2%) spect 4.15 (87%), 1.38 (4%), 0.90 (7%), 0.40 (2%) spect	γ_1 1.05, γ_2 1.7, γ_3 2.2, γ_4 2.75 (coin with γ_1), γ_5 3.3, γ_6 4.25, γ_7 4.8 ($\gamma_1/\gamma_2/\gamma_3/\gamma_4/\gamma_5$ / $\gamma_6/\gamma_7 \approx 3.7$ / 0.3/0.5/2.9/0.5/ 0.2/0.2) spect, γ - γ coin abs γ_1 1.04, γ_2 1.38, γ_3 1.90, γ_4 2.17, γ_5 2.40, γ_6 2.75, γ_7 3.24, γ_8 3.41, γ_9 3.78, γ_{10} 4.12, γ_{11} 4.33, γ_{12} 4.83 (γ_1 / $\gamma_2/\gamma_3/\gamma_4/\gamma_5/\gamma_6$ / $\gamma_7/\gamma_8/\gamma_9/\gamma_{10}/\gamma_{11}/$

Isotope Z A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
				Particles	Gamma transitions
⁶⁶ Ga (Cont.)					$\gamma_{12} \approx \sim 1.5/$ $\sim 0.2/0.14/0.24/$ $0.10/1.00/0.09/$ $0.14/0.08/0.07/$ $0.21/0.09$ scint spect, pair scint spect $1.06, 2.75, 3.25, 4.27,$ 4.8 scint spect $1.03, 2.75, 4.8$ spect, spect conv
⁶⁷ Ga		EC	77.9 h		$\gamma_1 0.090 (e_K/\gamma 0.074),$ $\gamma_2 0.092$ (with $Zn^{67m}, e_K/\gamma 0.63,$ $K/L 7), \gamma_3 0.182$ $(e_K/\gamma 0.011, K/L$ $\sim 12.5), \gamma_4 0.206$ $(e_K/\gamma 0.029, K/L$ $\sim 14), \gamma_5 0.296$ $(e_K/\gamma 0.0029), \gamma_6$ $0.388 (e_K/\gamma$ $0.0019), \gamma_7 0.496,$ $\gamma_8 0.790, \gamma_9 0.880$ $(\gamma_1/\gamma_2/\gamma_3/\gamma_4/\gamma_5/$ $\gamma_6/\gamma_7/\gamma_8/\gamma_9 \approx$ $0.073/1.00/0.81/$ $0.021/0.54/0.13/$ $0.004/0.002/$ $0.004)$ spect conv, scint spect, conv- $\gamma, \gamma-\gamma$ coinc
		No β^+ (lim 0.01%) No β^+	78.2 h		$\gamma_2 0.092 (e/\gamma 0.28,$ $K/L 6.5), \gamma_3 0.182$ $(e/\gamma 0.0091, K/L$ $9), \gamma_4 0.205 (e/\gamma$ $0.0055), \gamma_5 0.298$ $(e/\gamma 0.0025), \gamma_6$ $0.388 (e/\gamma 0.0017)$ $(\gamma_2/\gamma_3/\gamma_4/\gamma_5/\gamma_6$ $\approx 62/21/2/12/3)$ spect, spect conv, conv- γ coinc $\gamma_1 0.090 (4\%), \gamma_2$ 0.092 (with $Zn^{67m},$ $46\%, e/\gamma 0.6), \gamma_3$ $0.182 (21.5\%), \gamma_4$ $0.20 (2\%), \gamma_5$ $0.30 (12.5\%), \gamma_6$ $0.39 (4.5\%), \gamma_7$ $\sim 0.50 (<1\%),$ $\gamma_9 \sim 0.85 (<1\%),$ $\gamma_{10} \sim 1.05 (<1\%)$ scint spect, $\gamma-\gamma$ coinc
⁶⁸ Ga		$\beta^+ \sim 85\%,$ EC $\sim 15\%$	68 m	$1.88 (\sim 96\%),$ $0.77 (\sim 4\%)$ spect 2.0 abs 1.9 abs cl ch	1.10 spect

Isotope Z	A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
					Particles	Gamma transitions
Ga ⁶⁹		60.2				
³¹ Ga ⁷⁰			β^-	20.3 m 20 m	1.65 spect 1.62 abs ~1.6 cl ch	
Ga ⁷¹		39.8				
Ga ⁷²			β^-	14.3 h 14.1 h	3.15 (9%), 2.52 (8%), 1.5 (11%), 0.9 (32%), 0.6 (40%) spect 3.17 (8%), 2.57 (8%), 1.7 (?) (3%), 1.5 (?) (7%), 1.00 (26%), 0.74 (23%), 0.6 (?) (25%) spect	γ_1 2.491, γ_2 2.508 (γ_1/γ_2 0.63) spect 2.51 (26%), 2.21 (33%), 1.87 (8%), 1.59 (5%), 1.20 (?) (<2%), 1.05 (5%), 0.84 (100%), 0.68 (<2%), 0.63 (24%) spect, spect conv γ_1 γ_2 2.50 (15%), γ_3 2.18 (33%), γ_4 1.81 (10%), γ_5 1.57, γ_6 1.47, γ_7 1.30 ($\gamma_5 + \gamma_6 +$ γ_7 28%), γ_8 1.05 (15%), γ_9 0.835 (100%), γ_{10} 0.691 (5%), γ_{11} 0.631 (54%) spect, spect conv 3.4 (~0.03%), 3.1 (~0.1%) D- γ -p ion ch
Ga ⁷³			β^-	5.0 h	1.4 abs	0.0135, 0.054 spect conv
³² Ge ⁶⁶			β^+ (?)	~150 m		
Ge ⁶⁷			β^+	21 m		
Ge ⁶⁸			EC	250 d		
Ge ⁶⁹			EC ~67%, β^+ ~33%	39.6 h 39.7 h 40 h 37 h	1.215 (88%), 0.610 (10%), 0.22 (2%) spect	1.610, 1.340, 1.12, 0.870, 0.576, 0.388, 0.090 spect, β - γ coinc
Ge ⁷⁰		20.55				
Ge ⁷¹			EC, no β^+	11.4 d 11.3 d 11 d		No γ Ga K-x
Ge ^{72m}			IT	2.9×10^{-7} s delay coinc 5×10^{-7} s delay coinc		0.7 (e/γ very high) coinc abs conv
Ge ⁷²		27.37				
Ge ⁷³		7.67				
Ge ⁷⁴		36.74				

Isotope Z A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
				Particles	Gamma transitions
^{75m} Ge		IT	42 s 48 s	conv: 0.14 abs	0.175 scint spect ~0.150 scint spect
⁷⁶ Ge		β^-	82 m	1.137 (85%, not coinc with γ), 0.614 (15%)	0.265 (e/γ very small), 0.418,
			89 m	spect	0.572, other γ 's
			79 m	1.2 abs 1.3 abs	spect, spect conv, scint spect
					0.25 (~10%) No γ
⁷⁶ Ge	7.67				
^{76m} Ge		IT	~0.35 s		~0.06 scint spect
^{77m} Ge		β ~50%, IT	59 s	2.8 abs	0.38 scint spect
		~50% calc β^- , IT	57 s		
⁷⁷ Ge		β^-	12 h	2.196 (42%, coinc with γ), 1.379 (35%), 0.71 (23%) spect	0.042, 0.073 (coinc with 0.213 and 0.264 γ), 0.213, 0.264, 0.368, 0.418, 0.564, 1.105, 1.75 spect, β - γ , γ - γ coinc
					0.209, 0.268, 0.300, 0.327, 0.366, 0.408, 0.425, 0.466 spect conv
					0.6, 0.3 (coinc with β^-) abs, β - γ coinc
⁷⁸ Ge		β^-	2.1 h	~0.9 abs	γ
⁷⁰ As		β^+	52 m		
⁷¹ As		EC	60 h		0.162 spect conv
		EC, β^+	50 h		
⁷³ As		EC, β^+	26 h	3.34 (19%), 2.50 (62%),	0.835, others up to
			27 h	1.84 (12%), 0.67 (5%), 0.27 (2%) spect	3.0 (weak) spect
				2.8 abs	
⁷³ As		EC	76 d		0.0135 (L/M 5.4),
		No β^+	90 d		0.0539 ($K/L + M$ 5.6) spect conv, conv-conv coins 0.052 spect conv
⁷⁴ As		β^- 53%,	17.5 d	β^- :	γ_1 0.596, γ_2 0.635
		β^+ 47%	19.0 d	1.36 (51%), 0.69 (49%) spect	($\gamma_1/\gamma_2 = 4$) spect,
		β^- ~67%,	16 d	β - γ coinc	
		β^+ ~33%		1.45 (47%), 0.82 (53%) spect	0.593 spect
				β^+ :	0.582 spect
				1.53 (11%), 0.92 (89%) spect	
				0.96 spect	
⁷⁶ As	100				

Isotope Z A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
				Particles	Gamma transitions
⁵⁵ As ⁷⁶		β^- , no β^+ :	26.8 h	3.04 (60%), 2.49 (25%),	0.55, 1.20, 1.70 spect
		β^+ lim	26.3 h	1.29 (15%) spect	0.557, 1.22, 1.78
		0.03%	26.1 h	3.15 (54%), 2.57 (21%),	(weak) spect
		β^+ lum		1.4 (19%), 0.4 (7%)	0.558 (e_K/γ 0.002)
		0.07%		spect	spect conv
		β^+ lim		3.15 ($\sim 40\%$), 2.7	γ_1 0.57, γ_2 1.25, γ_3
		0.1%		($\sim 30\%$), 1.1	1.84, γ_4 2.15
				($\sim 30\%$) spect, β - γ	($\gamma_1/\gamma_2/\gamma_3/\gamma_4 =$
				coinc	1/0.4/weak/
					weak) spect
⁵⁷ As		β^-	38 h	0.700 spect	No γ
			40 h	0.679 spect No conv	
⁷⁸ As		β^-	90 m	4.1 ($\sim 70\%$), 1.4	0.27 abs
			80 m	($\sim 30\%$) abs	No γ coinc with
			65 m	1.4 cl ch	4.1 β^-
⁷⁸ As			~ 40 m		
⁷⁹ As		β^-	9 m	~ 2.1 abs	
			10 m		
⁷⁰ Se		β^+	~ 44 m		
⁷² Se		EC	9.7 d		
^{73(m²)} Se		β^+ , EC, IT 20% (?)	7.1 h	1.68 (1%), 1.318 (88%),	γ_1 0.0671 (conv in
			6.7 h	0.750 (10%), 0.25 (1%) spect	Se, K/L 7.6), γ_2 0.361 (K/L 8.6), γ_3 0.860, γ_4 1.310 ($\gamma_1/\gamma_2/\gamma_3/\gamma_4 =$ 10/163/11/8) spect, spect conv
⁷⁴ Se	0.87				
⁷⁵ Se		EC, no β^+	127 d		γ_1 0.067, γ_2 0.077,
			128 d		γ_3 0.098 (K/L
			115 d		~ 11 , $e/\gamma \sim 8$), γ_4
			120 d		0.124 ($e/\gamma \sim 0.3$), γ_5 0.138 (e/γ
					~ 0.1), γ_6 0.203, γ_7 0.269 (e/γ ~ 0.09), γ_8 0.281, γ_9 0.308, γ_{10} 0.405 ($e/\gamma \sim 0.00$) ($\gamma_2/\gamma_3/\gamma_4/\gamma_5/\gamma_6/\gamma_7/\gamma_8/\gamma_{10} \approx$ $0.14/\sim 0.007/\sim 0.02/\sim 0.05$

Isotope Z A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
				Particles	Gamma transitions
⁷⁶ Se (Cont.)					/0.14) spect, spect conv, γ - γ coinc 0.025, 0.066, 0.081, 0.097, 0.121, 0.136, 0.199, 0.265, 0.280, 0.305, 0.402 spect conv 0.076, 0.099 (?), 0.123, 0.137, 0.267, 0.283, 0.405 spect 0.098 (e/γ large), all other γ (e/γ very small) spect, spect conv
⁷⁶ Se	9.02				
^{77m} Se		IT	17.5 s		0.162 (K/L 4.6) spect conv 0.160 (e/γ large) spect, spect conv 0.165 spect conv abs
⁷⁷ Se	7.58				
⁷⁸ Se	23.52				
^{79m} Se		IT	3.5 m 3.9 m		0.096 (K/L 2.9) spect conv
⁷⁹ Se		β^-	$\leq 6.5 \times 10^4$ y sp act (est fission yield)	0.160 abs ~ 0.150 abs	No γ
⁸⁰ Se	49.82				
^{81m} Se		IT	56.5 m 57 m 59 m		0.103 (K/L 3.0) spect conv 0.104 (e/γ very large, K/L ~ 3.9) spect conv 0.099 (K/L ~ 4) spect conv
⁸¹ Se		β^-	17 m 18 m 19 m 13.6 m	1.38 spect 1.5 abs No conv	No γ
⁸² Se	9.19				
⁸² Se		β^-	67 s	3.4 abs No conv	
⁸³ Se		β^-	25 m 26 m 30 m	1.5 abs	0.950, 0.176, 0.061 (?), 0.04 (?) spect conv, scint spect 1.1, 0.37, 0.17 abs
⁸⁴ Se		β^-	~ 2 m		
⁸⁴ Br		β^+ , EC	~ 35 m		

Isotope Z A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
				Particles	Gamma transitions
$^{85}\text{Br}^{75}$		β^+ , <i>EC</i>	1.6 h 1.7 h	1.70 (46%), 0.8 (20%), 0.6 (15%), 0.3 (19%) spect ~1.8 abs	~0.6 abs
Br^{76}		β^+	17.2 h 16.5 h 15.7 h	3.57 (46%), 1.7 (10%), 1.1 (11%), 0.8 (14%), 0.6 (19%) spect 3.5 spect	1.2, 0.96, 0.75, 0.68, 0.42, 0.37, 0.33, 0.25 spect, spect conv ~2 abs
Br^{77}		<i>EC</i> 95%, β^+ 5%	57 h 58 h	0.336 spect 0.36 spect abs, spect	γ_1 0.160, γ_2 0.237, γ_3 0.284, γ_4 0.298, γ_5 0.520, γ_6 0.641, γ_7 0.813 ($\gamma_1/\gamma_2/\gamma_3/$ $\gamma_4/\gamma_5/\gamma_6/\gamma_7 =$ 0.6/20/0.2/0.2/ 100/8.6/25) spect, spect conv 0.160, 0.234, 0.299, 0.521 spect conv, α - γ coincide
Br^{78}		β^+	6.4 m 6.5 m	2.4 abs 2.3 abs	0.108, 0.046 spect conv
Br^{79}	80.52				
$\text{Br}^{80\text{m}}$		<i>IT</i>	4.58 h 4.5 h 4.4 h		0.049, 0.037 ($e/\gamma \sim 1.3$) ion ch 0.048, 0.036 spect conv 0.049, 0.037 spect conv 0.049 (L_I/L_{III} 1.0) spect conv 0.048 (e/γ very large), 0.037 ($e/\gamma \sim 1$) abs
Br^{80}		$\beta^- \sim 92\%$, β^+ $\sim 3\%$, <i>EC</i> $\sim 5\%$ β^+/β^- 0.037 $\beta^+ + EC$ $\frac{\beta^-}{\beta^+} 0.09$ β^+/β^- 0.028 β^+/β^- 0.03	18 m	β^- : 1.99 (85%), 1.1 (15%) spect 1.97 (80%), 1.1 (11%), 0.7 (9%) spect 1.99 spect β^+ : 0.87 spect 1.0 spect 0.73 abs	>0.6 abs <0.5 abs No γ
Br^{81}	49.48				
Br^{82}		β^- No <i>EC</i> or β^+ (lim 0.03%) No β^+ (lim 0.02%)	35.87 h 36.0 h 35.1 h 35.7 h	0.465 spect, β - γ coinc	γ_1 0.547, γ_2 0.608, γ_3 0.692, γ_4 0.766, γ_5 0.823, γ_6 1.031, γ_7 1.312 spect conv, spect

Isotope Z	A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
					Particles	Gamma transitions
³⁵ Br ⁸² (Cont.)						γ_1 0.535, γ_2 0.602, γ_3 0.750, γ_4 1.020, γ_5 1.292, γ_6 1.445 (γ_1/γ_4 / $\gamma_5/\gamma_7/\gamma_6 = 3.7$ / 3.5/1.0/0.85/0.40) spect 0.547, 0.615, 0.682, 0.752, 0.822, 1.026, 1.306, 1.453 spect 1.7-2.0 (0.14%) Be- γ -n reaction 0.07, 0.05 abs conv
Br ⁸³			β^-	2.33 h	0.940 spect	No γ
				2.4 h	0.94 spect	~ 0.046 (?) cl ch
Br ⁸⁴			β^-	30 m	4.68 (40%), 3.56 (9%),	0.89, 1.89 scint spect
				32 m	2.53 (16%), 1.72	
				33 m	(35%) spect	
Br ⁸⁶			β^-	3.00 m	2.5 abs	No γ
				3.0 m		
Br ⁸⁷		β^- , β^-n ($\sim 2\%$ of disinte- grations)		55.6 s (n)	β^- :	5.4, others ~ 3 abs,
				55.0 s (n)	2.6 (70%), 8.0 (30%)	abs sec, γ - γ coine
				56.1 s (β^-)	abs n (mean): 0.25 abs paraffin 0.30 p recoil in cl ch	
Br ⁸⁸			β^-	15.5 s		
Br ⁸⁹			β^- , β^-n	4.51 s (n)	n (mean):	
				4.45 s (n)	0.43 abs paraffin 0.65 p recoil in cl ch	
³⁶ Kr ⁷⁷			EC 70%, β^+ 30%	1.1 h	1.7 abs	γ
Kr ⁷⁸	0.354					
Kr ^{79m}			IT (?), no β^+	55 s		0.127 spect conv
Kr ⁷⁹			EC 95% (L/K 0.27), β^+ 5% EC (K) $\sim 90\%$, β^+ $\sim 10\%$ EC $\sim 98\%$, β^+ $\sim 2\%$	34.5 h	0.595 spect	0.263 ($e/\gamma \sim 0.02$),
				34 h	0.6 abs	0.044 spect conv
						0.2 abs
Kr ⁸⁰	2.27					
Kr ^{81m}			IT, no β^+	13 s		0.193 spect conv
				~ 10 s		0.187 spect conv
Kr ⁸¹			EC	2.1×10^5 y sp act		0.012 abs, Br K-x

Isotope <i>Z</i>	<i>A</i>	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
					Particles	Gamma transitions
⁸² Kr		11.56				
^{83m} Kr			<i>IT</i>	114 m 113 m		0.0322 (<i>e</i> / γ very large, <i>K/L + M</i> 0.35), 0.0093 (<i>e</i> / γ very large, <i>L/M</i> ~ 3) spect conv 0.032 (<i>K/L</i> 0.32), 0.009 (<i>e</i> / γ ~ 10) ion ch 0.046, 0.029 spect conv
⁸³ Kr		11.55				
⁸⁴ Kr		56.90				
^{86m} Kr			β^- 77%, <i>IT</i> 23%	4.36 h 4.4 h 4.5 h 4.6 h	0.855 spect 0.85 abs 0.95 abs 0.75 abs	0.150 (<i>e</i> / γ 0.057, coinc with β^-), 0.305 spect conv, β - γ coinc 0.17, 0.37 abs
⁸⁶ Kr			β^-	9.4 y ~ 10 y	0.695 (99 + %) , 0.15 (0.65%) spect, β - γ coinc abs 0.74 abs	0.54 (coinc with 0.15 β^-) scint spect, abs, β - γ coinc
⁸⁶ Kr		17.37				
⁸⁷ Kr			β^-	78 m 75 m 74 m	3.63 (75%), 1.27 (25%) spect	0.41, 1.89, ~ 2.3 scint spect
⁸⁸ Kr			β^-	2.77 h 2.8 h	2.8 (20%), 0.9 (12%), 0.52 (68%) spect 2.4 (weak), ~ 0.5 abs	0.028 (coinc with 0.5 β^- , <i>e</i> / γ ~ 0.1 , <i>K/L + M</i> 8) spect conv, β - γ coinc
⁸⁹ Kr			β^-	3.18 m 2.6 m 2.5 m	4.0 abs 3.9 calc from avg recoil energy	
⁹⁰ Kr			β^-	33 s	3.2 abs	
⁹¹ Kr			β^-	9.8 s 10 s ~ 6 s	~ 3.6 abs	
⁹² Kr			β^-	3.0 s		
⁹³ Kr			β^-	2.0 s		
⁹⁴ Kr			β^-	1.4 s		
⁹⁶ Kr			β^-	Short		
⁹⁷ Kr			β^-	~ 1 s		
⁸¹ Rb			<i>EC</i> 87%, β^+ 13%	4.7 h	0.990 spect	0.95 abs
^{82m} Rb			β^+	1.25 m	~ 3 abs; see ⁸² Sr	

Isotope Z A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
				Particles	Gamma transitions
$r\text{Rb}^{83}$		EC 94%, β^+ 6%	6.3 h	0.775 (76%), 0.175 (24%) spect 0.670 spect	0.188, 0.248, 0.322, 0.390, 0.423, 0.465, 0.558, 0.610, 0.690, 0.768, 0.818, 1.020, 1.314, 1.464 spect conv, spect
Rb^{83}		EC	83 d 107 d		~ 0.45 , ~ 0.15 spect conv
Rb^{84m}		IT , EC (weak) EC	23 m 20 m		γ_1 0.463 (not coin- with γ_2 or γ_3). γ_2 0.239, $\gamma_3 \sim 0.239$ (coin with γ_2). γ_4 0.890 ($\gamma_1/\gamma_4 \approx 7$) scint spect, $\gamma\gamma$ coinc
Rb^{84}		EC , β^+ , β^- (?) $EC/\beta^+ \sim 13$ $\beta^+/\beta^- \sim 6.2$	34 d 38 d	β^+ : 1.629 (39%), 0.822 (58%), 0.373 (?) (3%) spect 1.5 spect 1.3 abs	0.890 scint spect, spect conv 0.85 abs 0.8 abs
Rb^{85m}		IT	0.9×10^{-6} s delay coinc		0.513 spect, spect conv
Rb^{85}	72.15				
Rb^{86m}		IT , no EC EC	0.99 m 1.06 m		0.57 scint spect 0.78 abs
Rb^{86}		β^- No β^+ (lim 0.002%) No EC (lim 0.04%)	19.5 d	β_1 1.82 (80%), β_2 0.72 (20%) spect β_1 1.79 spect β_1 1.76, β_2 0.670 (coin- with γ) $\beta\gamma$ coinc spect β_1 1.82 ($\sim 67\%$), β_2 0.56 ($\sim 33\%$) abs β_2 0.72 (coinc with γ) $\beta\gamma$ coinc spect β_2 0.67 scint spect β_2 (12%) $\beta\gamma$ coinc	1.076 spect 1.081 spect 1.12 (coinc with β_2) coinc abs sec, $\beta\gamma$ coinc
Rb^{87}	27.85	β^-	6.0×10^{10} y sp act 6.4×10^{10} y sp act 6.2×10^{10} y sp act 7.6×10^{10} y 6.3×10^{10} y yield 5.8×10^{10} y sp act	0.275 spect, scint spect 0.270 scint spect	No γ

Isotope Z	A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
					Particles	Gamma transitions
⁸⁸ Rb			β^-	17.8 m	5.30 (78%), 3.6 (13%),	2.8, 1.86, 0.90 spect
				17.7 m	2.5 (9%) spect	γ_1 3.0, γ_2 1.7 (γ_1/γ_2
				17.5 m	5.13 (66%), 3.29	$\sim 1/10$) abs sec,
				18 m	(19%), 2.0 (15%)	β - γ coinc
					spect 5.20 ($\sim 66\%$), 3.6 ($\sim 17\%$), 1.8 ($\sim 17\%$) abs	
Rb ⁸⁹			β^-	15.4 m	4.5 abs	γ
				15.5 m	3.8 abs	
Rb ⁹⁰			β^-	2.74 m	5.7 abs	γ
Rb ⁹¹			β^-	1.67 m [Short]	4.6 abs	γ
Rb ⁹¹			β^-	14 m	3.0 abs	γ
Rb ⁹²			β^-	80 s [Short]		
Rb ⁹³			$[\beta^-]$	[Short]		
Rb ⁹⁴			$[\beta^-]$	[Short]		
Rb ⁹⁶			$[\beta^-]$	[Short]		
Rb ⁹⁷			$[\beta^-]$	[Short]		
⁸¹ Sr			EC, β^+	29 m	conv	
Sr ⁸²			EC, β^+ (?)	27 d	β^+ with Rb ^{82m}	0.95, ~ 0.40 , ~ 0.15
				25 d	3.15 spect	spect conv, abs
Sr ⁸³			EC, β^+	33 h	1.15 spect	0.040, 0.074, 0.101,
				38 h	1.35 abs	0.151, 0.165 spect conv
Sr ⁸⁴	0.56					
Sr ^{86m}			IT 86%, EC 14%	70 m		0.0075 (e/ γ very large), 0.150 (with EC), 0.225 (e/ γ 0.031, K/L ~ 5.5), 0.233 (e/ γ ~ 0.04) scint spect, spect conv, x- γ coinc 0.233, 0.152 spect conv
Sr ⁸⁶			EC No β^+	65 d		With Rb ^{86m} : 0.513 spect, spect conv 0.513 ($\sim 100\%$, e/ γ 0.008) spect conv, x- γ coinc, scint spect 0.514 ($\sim 100\%$, e/ γ 0.007, K/L, ~ 12) spect, spect conv, coinc abs
				66 d		

Isotope Z A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
				Particles	Gamma transitions
⁸⁶ Sr	9.86				
^{87m} Sr		IT	2.80 h 2.75 h		0.388 (K/L + M 5.79) spect conv 0.390 (e/γ 0.28, K/L + M 6.9) spect conv, x-conv coinc 0.388 (K/L + M 5.5) spect conv 0.388 spect conv 0.394 (K/L 7.2) spect conv 0.386 spect conv
⁸⁷ Sr	7.02				
⁸⁸ Sr	82.56				
⁸⁹ Sr		β ⁻	53 d 54 d 55 d	1.463 spect 1.48 spect 1.5 spect cl ch	No γ
⁹⁰ Sr		β ⁻	27.7 y	0.61 spect 0.54 spect 0.53 spect 0.6 abs	No γ
⁹¹ Sr		β ⁻	9.7 h 10 h	2.665 (26%), 2.03 (4%), 1.36 (29%), 1.09 (33%), 0.62 (7%) spect 3.2 (~60%), 1.3 (~40%) abs	0.551 (with Y ^{91m} , K/L + M 6.0), 0.64, 0.66, 0.747, 1.025, 1.413 (coinc with 0.630 γ) spect, scint spect, γ-γ coinc
⁹² Sr		β ⁻	2.7 h		
⁹³ Sr		β ⁻	7 m		
⁹⁴ Sr		β ⁻	~2 m		
⁹⁵ Sr		β ⁻	Short		
⁹⁷ Sr		β ⁻	Short		
⁹³ Y			70 m		
⁹³ Y			3.5 h		
⁹⁴ Y		β ⁺ , EC	3.7 h	2.0 abs	γ
⁹⁵ Y			5 h		
⁹⁶ Y		β ⁺	14.6 h	1.80 (~50%), 1.19 (~50%) spect	1.4 abs ~1.3 abs
^{87m} Y		IT No β ⁺	14 h		0.381 (K/L + M 5.41) spect conv 0.384 (e/γ 0.28) spect conv, ion ch, conv- x coinc 0.389 spect conv No γ > 1 Conv > 1

Isotope Z A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
				Particles	Gamma transitions
^{87}Y		EC 99+%, β^+ ~0.3% EC, β^+ (weak)	80.0 h 80 h	0.7 spect	0.485 (e/ γ 0.0035) spect conv, scint spect, γ - γ coinc 0.390 (with $\text{Sr}^{87\text{m}}$)
^{88}Y		EC EC 99+%, β^+ 0.19%	104 d 105 d	0.83 spect	0.908 (e/ γ 0.0003), 1.853 (e/ γ 0.0001), 2.76 spect conv, spect ~0.9 (e π / γ 0.00034), ~1.85 (e π / γ 0.00017) spect conv 0.908, 1.89 spect, γ - γ coinc 1.87 Be- γ -n reaction 2.8 (~1%) D- γ -n reaction
^{88}Y		β^+	2.0 h		
$^{89\text{m}}\text{Y}$		IT	~14 s		0.913 spect conv 0.917 spect conv ~0.9 (K/L + M 7.0) spect conv 0.92 (e/ γ 0.01) spect conv, scint spect
^{89}Y	100				
^{90}Y		β^-	61 h 62 h 65 h	2.18 spect 2.24 spect 2.25 spect 2.27 spect 2.2 spect, abs E (avg) 0.90 ion ch	No γ 1.4 (0.4%) abs
$^{91\text{m}}\text{Y}$		IT	51.0 m 50 m		0.551 (K/L + M 6.00) scint spect 0.61 (e/ γ ~0.1) abs, abs conv
^{91}Y		β^-	61 d 57 d	1.537 spect 1.54 spect 1.55 spect 1.56 spect	1.2, 0.2 (both <0.1%) abs, coinc abs sec, γ - γ coinc
^{92}Y		β^-	3.60 h 3.5 h	3.60, 2.7, 1.3 spect 3.5 abs 3.4 abs 3.6 abs	0.6 abs 0.7-1.1 abs
^{93}Y		β^-	10.0 h 11.5 h	3.1 abs	0.7 abs
^{94}Y		β^-	16.5 m 20 m	5.4 abs	1.4 abs
^{95}Y		β^-	10.5 m <1.5 h		
^{97}Y		β^-	Short		

Isotope Z	Per cent abundance A	Type of decay	Half-life	Energy of radiation, Mev	
				Particles	Gamma transitions
⁴⁰ Zr ⁸⁸		EC	~17 h genet		
Zr ⁸⁷		β^+ , EC	94 m 120 m	2.10 spect 2.0 abs	0.65, 0.35 abs
Zr ⁸⁸		EC	85 d		0.406 spect conv
Zr ^{89m}		IT, β^+ (weak)	4.4 m 4.5 m	2.5, ~0.85 (both weak) ($\beta^+/0.59 \gamma = 0.004$) spect, scint spect, β - γ coinc abs	0.586 (e/ γ 0.07, K/L + M 5.4) scint spect, spect conv 1.53 (~7%) scint spect
Zr ⁸⁹		EC ~75%, β^+ ~25%	79.3 h 77 h 78 h 80 h	0.910 spect 0.905 spect 0.890 spect	γ (with Y ^{89m}): 0.92 (e/ γ 0.01) spect conv, scint spect 0.913 spect conv 0.917 spect conv ~0.9 (K/L + M 7.0) spect conv
Zr ⁹⁰	51.46				
Zr ⁹¹	11.23				
Zr ⁹²	17.11				
Zr ⁹³		β^-	9.5×10^4 y sp act	0.063 scint spect	No γ
Zr ⁹⁴	17.40				
Zr ⁹⁶		β^-	65 d 66 d 63 d	0.371 (99%), 0.84 (1%) spect 0.39 (98%), ~1.0 (2%) spect 0.365 (95%), ~0.60 (weak), ~1.1 (weak) spect ~0.4 (98%), ~1.0 (2%) abs 0.40 abs 0.40, 0.88 β - γ coinc abs, abs	0.721 (e/ γ 0.0024), no 0.92 γ spect, spect conv 0.73, 0.92 (?) spect conv 0.88 abs 0.91 coinc abs sec Others
Zr ⁹⁶	2.80				
Zr ⁹⁷		β^-	17.0 h	1.91 spect 1.9 abs 2.2 abs 2.5 abs	With Nb ^{97m} : 0.747 (e/ γ 0.015) spect, spect conv, β - γ coinc, γ - γ coinc, β - conv coinc
Zr ^m		IT	0.83 s		~0.50 scint spect
⁴¹ Nb ⁹⁰		β^+	15:0 h 15 h 18 h	1.2 abs ~1.7 abs	0.14, 1.14, 2.23 scint spect 2.0 abs

Isotope Z A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
				Particles	Gamma transitions
^{91}Nb		<i>IT</i>	64 d 60 d		0.1045 (e/γ ~ 50 , K/L 2.1) spect conv, scint spect 0.105 ($K/L + M$ 2.1) spect conv Nb x
Nb^{91}		[<i>EC</i>]	Long		Zr x
Nb^{92}		<i>EC</i>	~ 13 h		2.35 scint spect
Nb^{92}		<i>EC</i> , no β^- (lim 0.05%) <i>EC</i> , no β^-	10.1 d 9.8 d 11 d	β^- (?): 1.3 cl ch 0.6 abs	0.930 (with <i>EC</i>) spect 0.933, 1.84 (weak) scint spect 1.0 abs 1.1 abs Zr- x
Nb^{92}		β^-	21.6 h		
Nb^{92m}		<i>IT</i>	3.65 y		0.0292 (e/γ very large, K/L 0.14) spect conv
Nb^{92}	100				
Nb^{94m}		<i>IT</i> 99+%, $\beta^- \sim 0.1\%$	6.6 m	1.3 abs	0.0415 (e/γ large, $K/L/M \approx 0.31$ / 1.00/0.36) spect conv 0.056 (e/γ large) abs conv 0.9 (weak) scint spect Nb K- x
Nb^{94}		[β^-]	$> 5 \times 10^4$ y yield $\gg 100$ y yield		
Nb^{96m}		<i>IT</i>	90 h 84 h		0.231 (e/γ very large) spect conv 0.232 ($K/L + M$ ~ 3.5) spect conv 0.216 (e/γ very large) spect conv Nb K- x
Nb^{96}		β^-	35 d 37 d	0.160 spect 0.159 spect 0.148 spect 0.146 spect 0.15 spect	0.745 (e/γ 0.0024) spect, spect conv 0.758 (e/γ 0.002) spect, spect conv 0.77 (e/γ 0.0016, K / $L \approx 4$) spect conv 0.77 spect, spect conv 0.75 spect conv

Isotope Z A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
				Particles	Gamma transitions
$_{41}\text{Nb}^{96}$		β^-	23.35 h 22.9 h	0.750 (92%), 0.37 (8%) spect	0.216 (7%, e/γ $< 23 \times 10^{-3}$),
				0.686 (92%), 0.37 (8%) spect	0.238 (10%, e/γ $< 16 \times 10^{-3}$),
				0.75 spect	0.451 (27%, e/γ 4×10^{-3}), 0.560 (61%, e/γ 1.7 $\times$ 10^{-3}), 0.770 (100%, e/γ 1.2 $\times$ 10^{-3}), 0.804 (6%, e/γ 1.3 $\times 10^{-3}$), 0.840 (16%, e/γ 1.2 $\times 10^{-3}$), 1.078 (52%, e/γ 0.5 $\times$ 10^{-3}), 1.187 (32%, e/γ 0.3 $\times 10^{-3}$) spect conv, spect
					0.455, 0.545, 0.745, 0.9, 1.05, 1.1 spect, spect conv
Nb^{97m}		IT	60 s		0.747 (e/γ 0.015, $K/L \geq 4$) spect, spect conv, β - γ coinc, γ - γ coinc, β -conv coinc
Nb^{97}		β^-	72.1 m	1.267 spect	0.665 (e/γ ~ 0.0015) spect, spect conv
			74 m	1.35 abs	
			75 m	1.4 abs	0.7 abs, β - γ coinc
Nb^{98}		β^-	30 m		
Nb^{99}		β^-	2.5 m	3.2 abs	
$_{42}\text{Mo}^{90}$		β^+ , EC	5.7 h genet		~ 0.1 , other γ 's abs
Mo^{91}		β^+	15.5 m	3.7 abs	No γ
			17 m	2.7 cl ch	
Mo^{91}		β^+	75 s	2.6 abs	0.3 abs
			73 s		
Mo^{92}	15.86				
Mo^{93m}		IT	6.95 h		0.262 (K/L 2.9), 0.69, 1.51 spect, spect conv
			6.75 h		0.256 (K/L 2.8), 0.7, 1.5 spect conv, scint spect 0.30 (e/γ 9), 0.70 (e/γ 0.005), 1.7 (e/γ ~ 0) abs, abs conv, spect conv, conv- γ coinc γ_1 0.29, γ_2 0.69, γ_3 1.46 ($\gamma_1/\gamma_2/\gamma_3 \approx$ 0.6/1/1) scint spect

Isotope Z	A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
					Particles	Gamma transitions
42Mo^{93}			EC	> 2 y		Nb K- γ
Mo^{94}	9.12					
Mo^{95}	15.70					
Mo^{96}	16.50					
Mo^{97}	9.45					
Mo^{98}	23.75					
Mo^{99}			β^-	67 h 68.3 h 63.5 h 64 h	1.23 (~80%), 0.45 (~20%), ~0.08 (weak) (?) spect 1.23 (87%), 0.54 (13%) spect 1.25, others, spect 1.2 abs	γ_1 0.040, γ_2 0.181 (K/L 5), γ_3 0.367, γ_4 0.741, γ_5 0.780 ($\gamma_3/\gamma_4/\gamma_5 \approx 10/$ 100/14) spect, spect conv 0.728, 0.360 (weak). 0.182 (weak) spect, β - γ , γ - γ coinc 0.179, 0.168 spect conv γ_4 0.745, γ_5 0.780, γ_6 0.850 ($\gamma_4/\gamma_5/\gamma_6 \approx 100/$ 50/30) spect With Tc^{99m} : 0.0018, 0.140, 0.142
Mo^{100}	9.62					
Mo^{101}			β^-	14.6 m 14 m	1.2, 2.1 abs ~1.0, 2.2 abs 1.9 abs	0.191 (K/L ~6), 0.960 (coinc with 1.2 β^-) spect conv, scint spect, β - γ , γ - γ coinc 0.15 scint spect
Mo^{102}			β^-	12 m 11 m		
Mo^{105}			β^-	~5 m		
43Tc^{92}			β^+ , EC	4.3 m	4.1 abs	1.3 abs
Tc^{92}			EC	43.5 m 47 m		0.389 spect, spect conv 1.50 abs conv
Tc^{93}			EC 93%, β^+ 7%	2.75 h 2.7 h	0.800 spect 0.83 abs	1.34 (coinc with β^+) scint spect, β - γ coinc abs 2.00 abs 2.4 abs
Tc^{94}			β^+ ~75%, EC ~25%	53 m 50 m	2.41 (coinc with γ) spect, β - γ coinc 2.5 abs	0.874 ($e/\gamma \sim 10^{-3}$), 1.85, 2.73, 3.27 spect, spect conv 0.9 abs

Isotope Z A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
				Particles	Gamma transitions
^{96m} Tc		EC 96+%,	60 d	0.4 cl ch	γ_1 0.810 (e/γ 0.001),
		IT ~3%, β^+ 0.2- 0.6%	52 d 62 d		γ_2 0.201 (e/γ 0.036), γ_3 0.570 (e/γ 0.002), γ_4 1.02 ($\gamma_1/\gamma_2/\gamma_3/\gamma_4 = 0.3/0.7/0.4/0.03$) spect, spect conv, γ - γ coinc 0.0390 spect conv 0.80, 0.20, 0.58, 1.03 scint spect 0.8, 0.2 abs
⁹⁶ Tc		EC	20.0 h		0.762 (~90%), 0.932 (~5%), 1.071 (~5%) spect conv, γ - γ coinc
		No β^+	20 h		0.76, 1.07, no 0.93 γ scint spect 0.78 abs
^{96m} Tc		IT	51.5 m		0.0344 (K/L 1.2) spect conv Tc K-x
⁹⁶ Tc		EC No β^+	4.20 d		γ_1 1.119 (e/γ 3×10^{-4}), γ_2 0.842 (e/γ 6×10^{-4})
			4.35 d		0.806 (e/γ 6×10^{-4}), γ_4 0.771 (e/γ 6×10^{-4}),
			4.2 d		γ_5 0.312 (K/L 6.4) ($\gamma_1/\gamma_2/\gamma_3/\gamma_4/\gamma_5 \approx 0.17/1.00/0.82/1.00/0.0012$) spect, spect conv, γ - γ coinc
			4.3 d		1.65, 1.89, 2.39 (?) (all weak) scint spect
^{97m} Tc		IT	90 d		0.0958 (K/L + M 1.6) spect conv
			91 d		0.097 (K/L ~2) spect conv
			95 d		0.097 (e/γ very large) abs conv 0.108 (e/γ large) abs conv
⁹⁷ Tc		[EC]	> 10 ⁴ y yield		
^{99m} Tc		IT	6.04 h		0.1403 (K/L 7.7), 0.1423 (~1%, K/L ~2.5) spect conv
			5.9 h		0.1412 (e/γ 0.11, K/L/M + N = 7.9/1/0.3), 0.0018 (e/γ very large)
			6.6 h		spect conv

Isotope Z A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
				Particles	Gamma transitions
^{99m} Tc (Cont.)					0.140 (K/L ~9) spect, spect conv 0.139 (K/L ~10) spect conv
⁹⁹ Tc		β^-	2.12 $\times 10^6$ y sp act 2.2 $\times 10^6$ y sp act	0.290 spect 0.292 spect 0.296 spect 0.30 spect	No γ
¹⁰⁰ Tc		β^-	15.8 s 17.5 s	2.8 abs 2.4 abs	0.55 scint spect
¹⁰¹ Tc		β^-	14.0 m 14.5 m 16.5 m	1.20 (>95%) abs; β - γ coinc 1.2 abs 1.3 abs	0.307 (coinc with β^- , K/L ~6) spect conv, β - γ coinc, scint spect 0.30 (coinc with β^-), 0.56 (weak) scint spect, β - γ coinc 0.26 scint spect No 0.56 γ
¹⁰² Tc		β^-	<25 s <1 m		
¹⁰⁶ Tc		β^-	Short		
¹⁰⁷ Tc		[β^-]	<1.5 m		
⁹⁴ Ru		EC	~57 m genet		
⁹⁶ Ru		EC, β^+	1.65 h	1.1 abs	1.0, 0.5 abs
⁹⁶ Ru	5.7				
⁹⁷ Ru		EC	2.8 d		0.217 spect 0.23 abs
⁹⁸ Ru	2.2				
⁹⁹ Ru	12.8				
¹⁰⁰ Ru	12.7				
¹⁰¹ Ru	17.0				
¹⁰² Ru	31.3				
¹⁰³ Ru		β^-	39.8 d 42 d 41 d 45 d	0.217 (~99%), 0.698 (~1%) spect 0.222 (94%), 0.684 (6%) spect 0.205 (strong), 0.670 (weak) spect 0.350 (50%), 0.665 (50%) spect 0.2 (92%), 0.7 (8%) abs, β - γ coinc	0.498 (coinc with 0.22 β^- , e/γ ~0.01) spect, spect conv, β - γ coinc 0.498 spect 0.494 (e_K/γ 0.006, K/L 6.5) spect, spect conv 0.499 (K/L 8.5), 0.611 (K/L 4), 0.295 (?), 0.053 (K/L 1.0) spect conv With Rh ^{103m} ; 0.040

Isotope Z A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
				Particles	Gamma transitions
Ru ¹⁰⁴	18.3				
⁴⁴ Ru ¹⁰⁴		β^-	4.5 h	1.150 spect	0.726 (coinc with β^-)
			4.4 h	1.15 spect	spect, β - γ coinc
				1.3 abs	0.75 abs 0.7 abs With Rh ^{105m} : 0.130
Ru ¹⁰⁶		β^-	1.0 y	0.0392 spect 0.038 spect	No γ
Ru ¹⁰⁷		β^-	4 m	~4 abs	
⁴⁵ Rh ⁹⁹		β^+	4.5 h	0.74 spect	
Rh ¹⁰⁰		EC ~95%, β^+ ~5%	19.4 h	3.0 spect	1.2 abs
			21 h		1.8 abs
Rh ¹⁰¹		EC	4.3 d		0.300, 0.148 spect
			5.9 d		conv 0.35 spect conv, abs
Rh ¹⁰³		β^- , β^+ EC (?)	210 d	β^- :	
			215 d	1.0 cl ch	
				1.1 abs	
Rh ^{103m}		IT		β^+ :	
				1.1 cl ch	
			57 m		
			56 m		0.0400 (K/L + M
			52 m		0.2) spect conv,
			45 m		β - γ coinc 0.0404 (e/γ very large) spect conv 0.0396 (K/L 0.1) spect conv 0.040 spect conv, abs conv 0.042 abs conv
Rh ¹⁰³	100				
Rh ^{104m}		IT	4.4 m		0.052 scint spect,
			4.7 m		scint spect, ion ch
			4.3 m		
Rh ¹⁰⁴		β^-	44 s	2.6 spect	0.04, 0.18, 0.95 abs,
				2.3 abs, cl ch	abs conv
				2.5 abs	
Rh ^{105m}		IT	45 s		0.130 (e/γ ~3, K/L
					1.5) scint spect,
					spect conv 0.127 spect conv
Rh ¹⁰⁶		β^-	36.5 h	0.570 (~96%), 0.25	0.322 (~10%, coinc
			37 h	(~4%) spect	with 0.26 β^-),
			34 h	0.57 spect	0.157 (very weak),
				0.26 (~10%) β - γ coinc	0.080 (?) scint
				abs	spect, β - γ coinc
					abs 0.320 (~3%) scint spect ~0.3 (~8%, not coinc with 0.6 β^-) abs, β - γ coinc 0.33 (weak) abs

Isotope <i>Z A</i>	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
				Particles	Gamma transitions
⁴⁵ Rh ¹⁰⁶		β^-	30 s 40 s	3.53 (68%), 3.1 (11%), 2.44 (12%), 2.03 (3%), others (6%) spect 3.55 (82%), 2.30 (18%) spect 3.5, ~ 2.3 abs, β - γ coinc	γ_1 0.513 (eK/γ 0.004, K/L 8), γ_2 0.624 (eK/γ 0.002), γ_3 0.87, γ_4 1.045, γ_5 1.55, γ_6 2.41 ($\gamma_1/\gamma_2/\gamma_3/\gamma_4/\gamma_5$ $\approx 100/53/3/8/$ 2.5/1) spect, scint spect, spect conv 0.510, 0.622 spect conv 0.51 (17%), 0.73 (17%), 1.25 ($\sim 1\%$) spect, β - γ , γ - γ coinc ~ 0.5 (eK/γ 0.005), ~ 0.7 (eK/γ <0.003) spect conv Others
Rh ¹⁰⁷		β^-	26 m 24 m	1.2 abs	
Rh ¹⁰⁹		[β^-]	<1 h		
⁴⁶ Pd ¹⁰⁰		EC	4.0 d		0.09, 1.8 abs
Pd ¹⁰¹		EC 90%, β^+ 10%	8 h 9 h	2.3 spect 0.5 abs	No γ
Pd ¹⁰²	0.8				
Pd ¹⁰³		EC	17.0 d		Rh K- x No γ With Rh ^{102m} : 0.040
Pd ¹⁰⁴	9.3				
Pd ^{105m}		IT	~ 23 s		0.20 ($e/\gamma \sim 0.4$)
Pd ¹⁰⁶	22.6				
Pd ¹⁰⁶	27.2				
Pd ¹⁰⁷		β^-	$\sim 7 \times 10^6$ y sp act	~ 0.04 abs	
Pd ¹⁰⁸	26.8				
Pd ^{109m}		IT	4.8 m		0.173 scint spect 0.160 ($e/\gamma \sim 0.6$)
Pd ¹⁰⁹		β^-	13.6 h 13.1 h 13 h 14.1 h	0.961 spect 0.95 spect	With Ag ^{109m} : 0.087 ($e/\gamma \geq 11$, $K/L + M$ 1.3) spect conv No γ
Pd ¹¹⁰	13.5				

Isotope Z A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
				Particles	Gamma transitions
^{111m}Pd		<i>IT</i> 75%, β^- 25%	5.5 h		0.16, 1.77 scint spect
Pd^{111}		β^-	22 m 26 m	2.13 spect 2.15 spect 3.5 abs	0.38, 0.56, 0.65, 0.73 scint spect
Pd^{112}		β^-	21 h	0.2 abs	No γ
^{102}Ag			16 m		
Ag^{104}		β^+ , <i>EC</i>	1.2 h		
Ag^{104}		β^+	27 m	2.70 spect	0.118, 0.556, 0.148 (?), 0.179 (?) spect conv
Ag^{106}		<i>EC</i>	40 d 45 d		0.0625 (e/γ very large, $K/L > 5$), 0.280 (coinc with 0.063 γ , $K/L 8$), 0.343 ($K/L 5.8$), 0.440 ($K/L 7$), weak γ 's: 0.154, 0.181, 0.391 spect conv, spect, $\gamma\text{-}\gamma$ coinc 0.064, 0.220 (wea 0.278, 0.340, 0.43, (weak) spect 0.281, 0.319, 0.333 0.345, 0.393, 0.417 spect conv
Ag^{106}		β^+ β^- (?) ~2%	24.0 m 24.5 m 24.4 m 24.3 m	β^+ : 1.95, 1.5 spect 2.0 abs 1.9 cl ch β^- (?): 0.45 spect Conv: ~0.5 (weak) spect conv	~0.5, >0.6 (wea spect conv, scint spect, abs No γ
Ag^{106}		<i>EC</i>	8.2 d 8.3 d 7.5 d		0.220, 0.409, 0.511 ($K/L 8$), 0.620, 0.717, 0.815, 1.04, 1.24, 1.55 spect conv 0.515, 0.722, 1.04, 1.54 spect 0.72, 1.06, 1.63 spect
Ag^{107m}		<i>IT</i>	44.3 s 44 s		0.094 (e/γ ~16, K/L 0.92) spect conv 0.093 spect conv
Ag^{107}	51.35				
Ag^{108}		β^- 98.5%, <i>EC</i> 1.5%	2.3 m 2.4 m	1.5 scint spect	0.45, 0.66 scint spect 0.43, 0.60 scint spect

Isotope Z A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
				Particles	Gamma transitions
^{109m}Ag		<i>IT</i>	39.2 s 40 s		0.0875 spect conv 0.087 ($e\gamma/\gamma \sim 6$, $K/L + M$ 0.75) spect conv 0.087 ($e/\gamma \geq 11$, $K/L + M$ 1.3) spect conv 0.087 spect conv
Ag^{109}	48.65				
Ag^{110m}		β^- , <i>IT</i> <i>IT</i> $\geq 3\%$ No <i>EC</i> (lim 3%) No β^+ : lim β^+/β^- 0.002% lim β^+/β^- 0.05%	270 d 225 d	With Ag^{110m} and Ag^{110} : 0.087 ($\sim 58\%$), 0.530 ($\sim 35\%$), 2.12 ($\sim 3\%$), 2.86 ($\sim 3\%$), others (?) spect 0.088 (65%), 0.520 (33%), 2.89 ($\sim 2\%$) spect 0.590, 2.24, 2.91 spect 0.09 (coinc with γ), 0.57 (coinc with γ), 0.19 (?) abs, β - γ coinc 0.09, 0.59 cl cb 0.59 spect	0.116 (e/γ very large: K/L 1.3), 0.656 (e/γ 0.0025), 0.676, 0.706, 0.759, 0.814, 0.885, 0.935, 1.389, 1.516 spect conv, spect, β - γ coinc, γ - γ coinc 0.116 (conv in Ag), 0.438, 0.446, 0.471, 0.499, 0.542, 0.575, 0.619, 0.657, 0.677, 0.705, 0.723, 0.764, 0.817, 0.884, 0.937, 1.384, 1.504 spect, spect conv 0.116 ($K/L + M$ 1.8), 0.447, 0.618, 0.655 ($K/L + M$ 4.3), 0.687, 0.706 ($K/L + M$ 6.6), 0.740, 0.759 ($K/L + M$ 6.5), 0.815 ($K/L + M$ 4.1), 0.883 ($K/L + M$ 4.2), 0.932 ($K/L + M$ 6.5), 1.386 ($K/L + M$ 6.5), 1.480, 1.506 ($K/L + M$ 4.5) spect conv 0.666 ($K/L + M$ 14) spect conv 0.66, 0.90, 1.40 spect conv, spect ~ 0.7 , ~ 0.9 (coinc with 0.7 γ) γ - γ coinc abs 1.7 $< \gamma < 2.2$ Be- γ - π reaction Others
Ag^{110}		β^-	24.2 s 22 s	2.24 ($\sim 60\%$), 2.82 ($\sim 40\%$) scint spect	0.66, ~ 0.9 (weak) scint spect

Isotope Z A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
				Particles	Gamma transitions
^{111m}Ag			[< 5 m]		
Ag^{111}		β^-	7.6 d 7.5 d	1.04 (91%), 0.80 (1%), 0.70 (8%) spect 1.06 spect 1.1 (93.5%), 0.7 (6.5%) abs	γ_1 0.243 (e/ γ < 0.08), γ_2 0.340 (e/ γ ~ 0.015) (γ_2/γ_1 ~ 8) spect, spect conv, β - γ coin., γ - γ coin. 0.33 (~6.5%) abs
Ag^{112}		β^-	3.20 h 3.2 h	4.2 scint spect 3.6 abs 3.5 abs	0.625, 1.40 0.86 abs No γ coin with 3.5 β^-
Ag^{113}		β^-	5.3 h	2.0 scint spect 2.1 abs 2.2 abs	No γ
Ag^{114}		β^-	2 m 3 m	Hard β^-	
Ag^{115}		β^-	20 m 21 m genet 22 m	~ 3 abs	No γ
^{104}Cd		β^+	59 m	0.93 spect	0.0666 (K/L + M ~ 10), 0.0835 (K/ L + M ~ 15), 0.124, 0.134, other γ 's spect conv
Cd^{105}		EC, β^+	57 m 55 m 65 m	1.68 spect 1.5 abs	0.0255, 0.0494, 0.0525, 0.262, 0.293, 0.308, 0.312, 0.317, 0.321, 0.341, 0.347, 0.433, 2.1 spect conv, scint spect
Cd^{106}	1.215				
Cd^{107}		EC 99+%, β^+ 0.31%	6.7 h	0.32 spect	0.846 (0.4%, e/ γ ~ 10 ⁻³) spect, spect conv 0.7 abs With Ag^{107m} : 0.094
Cd^{108}	0.875				
Cd^{109}		EC (L/K 0.28) EC No β^+	470 d 330 d		With Ag^{109m} : 0.0875 spect conv 0.087 (eK/ γ ~ 6, K/L + M 0.75) spect conv 0.087 (e/ γ ≥ 11, K/L + M 1.3) spect conv

Isotope Z A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
				Particles	Gamma transitions
⁴⁸ Cd ¹¹⁰	12.39				
Cd ^{111m₂}		IT	48.6 m 48.7 m		γ_1 0.150 ($e/\gamma \sim 3$, K/L 2.0), γ_2 0.246 (e/γ 0.064, K/L 5.1) spect conv γ_1 (e/γ 2.3) γ_1 0.149, γ_2 0.247 scint spect γ_1 0.146, γ_2 0.235 spect conv
Cd ^{111m₁}		IT	8×10^{-8} s delay coinc 10×10^{-8} s delay coinc		0.247 scint spect
Cd ¹¹¹	12.75				
Cd ¹¹²	24.07				
Cd ^{113m}		β^-	5.1 y	0.59 scint spect 0.5 abs	
Cd ¹¹³	12.26				
Cd ¹¹⁴	28.86				
Cd ^{115m}		β^-	43 d 44 d	1.61 ($\sim 98\%$), 0.7 ($\sim 2\%$), ~ 0.3 (weak) spect 1.5 abs 1.4 abs; 0.4 (coinc with γ , $\sim 1\%$) β - γ coinc abs 1.7 abs ~ 0.8 ($\sim 1.4\%$) abs	0.46, 0.50, 0.96, 1.28 scint spect, γ - γ coinc γ_1 0.48, γ_2 0.94 (coinc with γ_1), γ_3 1.30 (not coinc with γ_1 or γ_2) (γ_1 / $\gamma_2/\gamma_3 \approx 13/100$ / 40) scint spect, γ - γ coinc
Cd ¹¹⁶		β^-	53 h 54 h 56 h	1.11 (58%), 0.58 (42%) spect 1.11 ($\sim 60\%$), 0.59 ($\sim 40\%$) spect ~ 1 ($\sim 85\%$), ~ 0.5 ($\sim 15\%$) β - γ coinc abs	0.335 (with In ^{115m}), 0.360, 0.500, 0.525 scint spect, γ - γ coinc 0.522 spect 0.336, 0.344, 0.349, 0.363 (?), 0.369, 0.424, 0.452, 0.525, 0.559, 0.713 spect conv With In ^{115m} . 0.334
Cd ¹¹⁶	7.58				
Cd ^{117m}		IT	3.0 h 2.9 h 2.8 h 2.7 h		With Cd ¹¹⁷ : ~ 1.2 abs
Cd ¹¹⁷		β^-	~ 50 m	~ 1.6 , ~ 3.0 abs	
⁴⁹ In ¹⁰⁷		β^+	30 m 33 m	~ 2 spect	

Isotope Z A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
				Particles	Gamma transitions
^{108}In		β^+	50 m ~55 m	2.31 spect 2.2 abs ~2 spect	0.285 (<5%, K/L ≥ 3) spect conv
^{109}In		β^+, EC	4.3 h 4.2 h 6.5 h 5.2 h	0.75 abs ~2 (weak)	0.427, 0.347, 0.205 (K/L 3), 0.058 (K/L 0.9) spect conv
^{110m}In		EC 99+%, IT ~0.3%	5.0 h 4.9 h ~5 h		0.119 (with IT, K/L 4.5), 0.935, 0.885, 0.661 spect conv 0.119 (weak), 0.937, 0.887, 0.654 spect conv
^{110}In		β^+, EC	66 m 65 m	2.25 spect	0.654 spect conv 0.677 (e/γ 0.005) spect, spect conv
^{111}In		EC No β^+ (lim 0.06%) No β^+	2.84 d 2.7 d		0.172 (~100%, e/γ 0.12, K/L 6.6), 0.247 (~100%, e/γ 0.064, K/L 5.19), 0.330 (weak), 0.093 (weak) spect, spect conv, conv-conv coinc abs 0.173 (e/γ 0.09, K/L ~8), 0.247 (e/γ 0.04, K/L ~5) γ -conv, conv- conv, γ - γ coinc 0.171 (K/L + M 7.03), 0.246 (K/L + M 4.79) spect conv 0.173 (K/L + M 6.6), 0.247 (K/L + M 5.3) spect conv 0.173 (K/L 6.6), 0.247 (K/L 5.4) spect conv 0.25 (coinc with 0.17 γ) scint spect, γ - γ delay coinc
^{112m}In		IT	20.9 m 20 m 23 m		0.154 (e/γ >4) spect conv 0.16 spect conv 0.16 (e/γ large) abs
^{112}In		β^-, β^+, EC $\beta^-/\beta^+ 2.7$	14.5 m	β^+ : 1.74 spect 1.7 spect β^- : 0.67 spect	

Isotope Z A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
				Particles	Gamma transitions
$^{115}\text{In}^m$		<i>IT</i>	2.5 s		0.153 scint spect
^{115}In		<i>IT</i>	104 m 105 m		0.392 (<i>K/L</i> + <i>M</i> 4.21) spect conv 0.393 spect conv 0.39 spect conv 0.39 (<i>K/L</i> 5.4) spect conv 0.39 (<i>e/γ</i> 0.55) scint spect ~0.39 (<i>e/γ</i> 0.35, <i>K/L</i> 5.4) spect conv, ion ch
^{113}In	4.23				
^{114m}In		<i>IT</i> , no <i>EC</i>	49 d		γ_1 with ^{114m}In , γ_2 , γ_3 , γ_4 , γ_5 with ^{114}In γ_1 0.190, γ_2 0.552, γ_3 0.722, γ_4 1.27 (γ_1 / $\gamma_2/\gamma_3/\gamma_4 \approx 100$ / 18.6/18.6/1.2) spect γ_1 0.192 (<i>e/γ</i> 4.2, <i>K/L</i> 1.10) spect conv, scint spect γ_1 0.190 (<i>K/L/M</i> $\approx$ 1.18/1.00/0.18) spect conv γ_1 0.191 spect conv γ_1 (<i>K/L</i> 1.30) spect conv γ_1 0.192 (<i>e/γ</i> 4, <i>K/L</i> 1.1), γ_2 0.55, γ_3 0.72 spect conv. γ_1 (<i>e/γ</i> 4) spect conv γ_1 (<i>K/L</i> 1.16) spect conv $\gamma_4/\gamma_2 \sim 0.06$ scint spect, γ - γ coinc γ_5 0.576, γ_6 1.30 (coinc with γ_2) spect, γ - γ coinc
^{114}In		$\beta^- \leq 97\%$, <i>EC</i> $\geq 3\%$, β^+ ~0.01% $\beta^- 99+\%$, β^+ 0.015% $\beta^- 99+\%$, β^+ ~0.01%	72 s	β^- : 1.984 spect 1.98 spect 2.01 spect 2.05 abs β^+ : ~1 abs 0.65 spect	
^{115m}In		<i>IT</i> , β^- <i>IT</i> 95%, β^- 5%	4.50 h 4.53 h	0.83 spect	0.335 (<i>e/γ</i> 0.98, <i>K/L</i> + <i>M</i> 3.76) spect conv

Isotope Z A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
				Particles	Gamma transitions
⁴⁹ In ^{115m} (Cont.)					0.338 ($e/\gamma \sim 1$, K/L 5.0, $K/L + M$ 4.0) spect conv 0.335 (K/L 5.3) spect conv 0.337 spect, spect conv 0.336 spect
In ¹¹⁵	95.77	β^-	6×10^{14} y sp act $\sim 10^{14}$ y sp act	0.63 abs	
In ^{116m}		β^-	53.93 m 54.31 m 54.05 m	1.00 (51%), 0.87 (28%), 0.60 (21%) spect, β - γ coinc 0.85 spect, el ch 0.7 β - γ coinc abs	2.090 (25%), 1.487 (21%), 1.274 (75%, e/γ 5.7×10^{-4}), 1.085 (54%, e/γ 8.4×10^{-4}), 0.406 (25%), 0.137 (3%) spect conv 0.137, 0.171 spect conv
In ¹¹⁶		β^-	13 s	2.95 abs 2.8 el ch	No γ
In ^{117m}		IT, β^-	~ 70 m	See In ¹¹⁷	
In ¹¹⁷		β^-	~ 2.5 h 1.95 h 1.90 h	With In ^{117m} (?): 1.726 spect 1.73 spect 1.95 abs With In ¹¹⁷ : 0.7 abs	0.161, 0.558 scint spect
In ¹¹⁸		β^-	4.5 m	1.5 abs	γ
In ¹¹⁹		β^-	17.5 m	2.7 abs	No γ
⁵⁰ Sn ¹⁰⁸		EC	4.0 h 4.5 h		
Sn ¹¹¹		EC $\sim 71\%$, β^+ $\sim 29\%$	35.0 m 35 m	1.51 spect 1.5 abs	
Sn ¹¹²	0.95				
Sn ¹¹³		EC, no β^+ EC (L/K ~ 0.8)	112 d 118 d 105 d		With In ^{113m} : 0.393 0.401 (weak), 0.255 (weak) spect conv No 0.09 γ
Sn ¹¹⁴	0.65				
Sn ¹¹⁵	0.34				
Sn ¹¹⁶	14.24				

Isotope Z A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
				Particles	Gamma transitions
$_{50}\text{Sn}^{117\text{m}}$		<i>IT</i>	14.0 d 15 d		0.159 (e/γ very large, K/L 2.2), 0.162 (eK/γ 0.10) spect conv, $e-\gamma$ coine, $x-\gamma$ coine 0.156 (e/γ large, $K/L \sim 7$), 0.159 spect conv 0.157 (K/L 2.2) spect conv 0.152 (K/L 2.4) spect conv
Sn^{117}	7.57				
Sn^{118}	24.01				
$\text{Sn}^{119\text{m}}$		<i>IT</i>	~ 250 d ~ 245 d		γ_1 0.0653 (K/L 0.51, $L/M \sim 4$), γ_2 0.0242 ($L/M \sim 4$) spect conv γ_2 0.0238 ($e/\gamma \sim 7$) scint spect, ion ch, $\gamma-\gamma$ coine γ_1 0.065, γ_2 0.024 scint spect, ion ch conv γ_1 0.064 (K/L 0.82) spect conv
Sn^{119}	8.58				
Sn^{120}	32.97				
$\text{Sn}^{121\text{m}}$		β^-	> 400 d	0.42 spect	
Sn^{121}		β^-	27.5 h 28 h ~ 27 h	0.383 spect 0.4 abs 0.35 abs	No γ
Sn^{122}	4.71				
Sn^{123}			39.5 m 40 m 41.5 m 39 m	1.26 spect 1.3 abs 1.1 abs	0.153 spect conv, $\beta-e$ coine 0.153 scint spect
Sn^{123}			136 d 125 d 130 d 126 d	1.42 spect 1.3 abs	No γ
Sn^{124}	5.98		Double β^- decay: (lower limit, by sp act) $> 2 \times 10^{17}$ y $> 10^{17}$ y $> 10^{16}$ y		

Isotope Z A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
				Particles	Gamma transitions
$^{125}_{50}\text{Sn}$		β^-	9.5 m 9.8 m	2.04, 1.17, 0.51 (?) spect 2.06, ~ 0.5 abs	0.326, others > 1 (weak) spect, spect conv 0.38 (coinc with 2.06 β) abs, β - γ coins 1.37 (weak) scint spect
$^{126}_{50}\text{Sn}$		β^-	9.4 d 10.0 d 9.5 d	2.37 ($\sim 95\%$), 0.40 ($\sim 5\%$) spect 2.33 spect ~ 0.5 (10%) β - γ coinc abs	1.90 scint spect, abs 1.67 coinc abs sec
$^{128}_{50}\text{Sn}$		β^-	~ 50 m yield		
$^{127}_{50}\text{Sn}$		β^-	1.5 h yield		
$^{116}_{51}\text{Sb}$		β^+	60 m	~ 1.45 spect	
$^{117}_{51}\text{Sb}$		EC	2.8 h		0.156 spect conv
$^{118}_{51}\text{Sb}$		β^+	3.5 m 3.6 m	3.1 abs, spect	
$^{118}_{51}\text{Sb}$		EC	5.1 h	$e \sim 0.2$ abs	0.260 spect conv 1.5 abs
$^{119}_{51}\text{Sb}$		EC	39 h		No γ Sn K-x
$^{120}_{51}\text{Sb}$		EC	6.0 d		~ 1.1 abs
$^{120}_{51}\text{Sb}$		β^+ , EC	16.4 m 16.6 m 17 m	1.70 spect	γ_1 0.90, γ_2 1.30, γ_3 2.20 ($\gamma_1/\gamma_2/\gamma_3$ $\sim 0.08/0.35/0.01$) (e/γ very small) spect
$^{121}_{51}\text{Sb}$	57.25				
$^{122m}_{51}\text{Sb}$		IT	3.5 m		0.068 scint spect 0.059, 0.074 ion ch
$^{122}_{51}\text{Sb}$		β^-	2.80 d 2.8 d	β_2 1.46 (coinc with γ) β - γ coinc spect β_1 1.94, β_2 1.36 spect β_1 1.8, β_2 1.2 abs, coinc abs β_1 1.8 abs	0.568 spect 0.57 spect conv 0.56 (e_K/γ 0.0049), 0.68
$^{123}_{51}\text{Sb}$	42.75				
$^{124m}_{51}\text{Sb}$		IT, β^-	21 m		0.0185 (e/γ very large)
$^{124m}_{51}\text{Sb}$		IT, β^-	1.3 m	3.2 abs	0.012 (e/γ very large)

Isotope Z A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
				Particles	Gamma transitions
$_{51}\text{Sb}^{124}$		β^- No β^+ , no EC	60 d	β_1 2.291 (21%), β_2 1.69 (7%), β_3 0.95 (7%), β_4 0.68 (26%), β_5 0.50 (39%) spect β_1 2.37 (21%), β_2 1.62 (8%), β_3 1.00 (9%), β_4 0.65 (44%), β_5 0.48 (18%) spect β_4 0.654 spect	0.121, 0.607, 0.653, 0.730, 1.708, 2.04 spect, spect conv 0.61 γ (40-50%, eK/γ 0.0043) spect, spect conv 0.61 γ (eK/γ 0.0036) spect conv 0.603 ($\sim 100\%$, e/γ ~ 0.002), 0.650, 0.714, 1.708, 2.06 spect, spect conv 0.598, 0.645, 0.817, 1.67, 2.07 spect
Sb^{125}		β^-	~ 2.7 y	0.616 (18%), 0.299 (49%), 0.128 (33%) spect 0.621, 0.288, others (?) spect	0.637, 0.601, 0.465, 0.425, 0.175, 0.035 spect, spect conv, coinc 0.646 (weak, e/γ very small), 0.809 (strong), 0.466 (weak, e/γ very small), 0.431 (strong), 0.174 (strong), 0.125 (weak) spect, spect conv, coinc With Te^{125m} : 0.110, 0.035
Sb^{126}		β^-	9 h	~ 1 abs	0.90, ~ 0.4 (both coinc with β^-) scint spect, β - γ coinc
Sb		β^-	10 m		
$\text{Sb} \sim^{126}$		β^-	28 d ~ 30 d	1.9	
Sb^{127}		β^-	93 h 95 h	1.2 abs 0.8 abs	0.72 abs
Sb^{129}		β^-	4.2 h		
Sb^{130}		β^-	40 m		
Sb^{130}		β^-	12 m 10 m		
Sb^{131}		β^-	23.1 m ~ 20 m		
Sb^{132}		β^-	2 m ~ 5 m ~ 2 m		
Sb^{133}		β^-	4.4 m 4.2 m		

Isotope Z A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
				Particles	Gamma transitions
$_{51}\text{Sb}^{134,135}$		β^-	~ 50 s 45 s		
$_{52}\text{Te}^{<118}$		β^+	2.5 h		
Te^{118}		EC	6.0 d		No γ (?)
Te^{119}		EC	4.5 d	Conv: 0.2, 0.5 spect conv	1.6 abs
Te^{120}	0.089				
$\text{Te}^{131\text{m}}$		IT	154 d 143 d 125 d 140 d		γ_1 0.082 (e/γ very large, K/L 0.75), γ_2 0.213 (e_K/γ 0.09, K/L 7.3) (γ_2 coinc with γ_1) spect conv, conv- γ coinc, conv-conv coinc γ_1 0.0818, γ_2 0.214 spect conv
Te^{131}		EC	17 d ~ 16 d		γ_1 0.506 (13%, e/γ ~ 0.018 , $K/L +$ M 6), γ_2 0.573 (87%, not coinc with γ_1 , e/γ 0.009, $K/L + M$ 6) spect conv, scint spect 0.575 spect conv ~ 0.61 (e_K/γ 0.004) spect conv 0.6 abs
Te^{132}	2.46				
$\text{Te}^{133\text{m}}$		IT	104 d 121 d		γ_1 0.0885 (e/γ very large, K/L 0.68), γ_2 0.159 (e_K/γ 0.18, K/L 8.9) (γ_1 coinc with γ_2) spect conv, γ -conv coinc, conv-conv coinc, abs γ_1 0.0887, γ_2 0.159 spect conv γ_1 (L_1/L_{III} 0.5) spect conv γ_2 (e_K/γ 0.19) scint spect No 0.25 γ (lim 0.5%) scint spect Others
Te^{133}	0.87				
Te^{134}	4.61				

Isotope Z A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
				Particles	Gamma transitions
^{126m}Te		<i>IT</i>	58 d		0.110 (eK/γ ~ 160 , $K/L + M$ 1.15), 0.0355 (eK/γ ~ 11.7 , $K/L/M \approx$ 7.3/1.0/0.18) spect conv, ion ch, scint spect, $x\text{-}\gamma$ coinc, $x\text{-}x$ coinc 0.110, 0.0353 spect conv 0.110 ($K/L + M$ 1.1), 0.035 spect conv 0.110 (K/L 1.2) spect conv 0.109 ($e/\gamma > 100$, $K/L \sim 1.5$, L/M ~ 3.5) spect conv
Te^{128}	6.99				
Te^{126}	18.71				
Te^{127m}		<i>IT</i>	115 d 90 d		0.0885 (K/L 0.75) spect conv 0.0887 spect conv 0.086 (e/γ very large, K/L 0.75) spect conv
Te^{127}		β^-	9.3 h	0.7 abs	No γ
Te^{128}	31.79				
Te^{129m}		<i>IT</i>	33.5 d 32 d		0.1060 spect conv 0.106 ($K/L \sim 1$) spect conv 0.102 (e/γ very large, $K/L \sim 1$) spect conv
Te^{129}		β^-	72 m 70 m 67 m	1.8 spect 1.7 abs	0.3, 0.8 abs
Te^{130}	34.49		Double beta decay: $\sim 10^{11}$ y Xe ratios, mass spect		
Te^{131m}		<i>IT</i>	30 h		0.177 (K/L 2) spect conv
Te^{131}		β^-	24.8 m 25 m	2.0 ($\sim 55\%$), 1.4 ($\sim 45\%$) abs, $\beta\text{-}\gamma$ coinc	0.16, 0.7 abs
Te^{132}		β^-	77.7 h 77 h	0.22 spect 0.3, ~ 0.1 (?) abs ~ 0.3 abs	0.231 scint spect 0.22 abs

Isotope Z A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
				Particles	Gamma transitions
^{128m} Te		IT	63 m 60 m		~0.4 scint spect With ¹²⁸ Te: 0.6, 1.0 abs
¹²⁹ Te		β^-	2 m	2.4 (~30%), 1.3 (~70%) abs	0.6, 1.0 abs
¹³⁴ Te		β^-	44 m 43 m		
¹³⁵ Te		β^-	<2 m		
Te		β^-	~1m		
¹²⁰ I		β^+	30 m	4.0 abs, spect	
¹³¹ I		β^+	1.5 h 1.8 h	1.2 abs, spect 1.2, 4.0 (weak) Conv: 0.185 spect	
¹³² I		β^+	3.6 m 3.4 m 4 m	2.9 abs 3.1 abs	
¹³³ I		EC	13.0 h 13 h		0.159 spect, spect conv
¹³⁴ I		EC ~70%, β^+ ~30%	4.5 d 4.0 d	2.20 (51%), 1.50 (44%), 0.7 (5%) spect 2.1 spect, abs	0.603, 0.73, 1.72, 1.95 spect, spect conv No γ coin with 2.2 β^+ , β - γ coin γ - γ , β - γ coin
¹³⁵ I		EC (L/K 0.23) EC (L/K 0.3) No β^+	60.0 d 56 d		0.035 ion ch 0.0355, no 0.109 γ spect conv
¹³⁶ I		EC ~58%, β^- ~40%, β^+ (?) ~2%	13.0 d 13 d	β^- : 1.268 (27%), 0.85 (73%) spect 1.24 (~25%), 0.85 (~75%) spect 0.865 spect	With β^- : 0.382 spect conv, β - γ coin 0.395 spect, spect conv With EC: 0.64 (weak) scint spect, α - γ coin, γ - γ coin
¹³⁷ I	100				
¹³⁸ I		β^- 95.0%, EC + β^+ 5.0% EC/ β^- 0.063	24.99 m	2.02 spect	0.428 (7%) spect
¹³⁹ I		β^-	1.72 $\times 10^7$ y sp act 3 $\times 10^7$ y sp act	0.12 scint spect 0.12 abs 0.12 ion ch 0.13 abs	0.039 (coin with β^- , eK/γ ~6, K/L ~40) ion ch, β - γ coin

Isotope Z A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
				Particles	Gamma transitions
^{130}I		β^-	12.6 h	1.03 (~60%), 0.61 (~40%) spect	0.744 (e_K/γ 0.003), 0.667 (e_K/γ 0.004), 0.537 (e_K/γ 0.007), 0.417 (coinc with 0.6 β^- , e_K/γ 0.012) (e_K >> e_L for all γ 's) spect, spect conv, β - γ , γ - γ coinc
^{131}I		β^-	8.141 d 8.05 d 8.16 d 8.04 d	0.815 (0.7%), 0.608 (87.2%), 0.335 (9.3%), 0.250 (2.8%) spect, β - γ coinc 0.810, 0.606, 0.335, 0.250 spect, β - γ coinc 0.807, 0.606, 0.339 spect E (avg) 0.189 ion ch	0.080 (2.2%, coinc with 0.284 γ , e_K/γ 1.73, K/L 7), 0.163 (coinc with Xe^{131m2}), 0.284 (5.3%, coinc with 0.608 β^- , e_K/γ 0.047, K/L 5), 0.364 (80%, coinc with 0.608 β^- , e_K/γ 0.018, K/L 8), 0.637 (9%, coinc with 0.335 β^- , e_K/γ 0.0037, K/L 9), 0.722 (3%, coinc with 0.250 β^- , e_K/γ 0.0028, K/L 8) spect, spect conv, β - γ delay coinc, scint spect γ_1 0.080133, γ_2 0.28413, γ_3 0.36418 ($\gamma_1/\gamma_2/\gamma_3 \approx 5/9/$ 100) cryst spect γ_3 0.284 (e_K/γ 0.052, K/L 3.3), γ_3 0.364 (e_K/γ 0.021, K/L 5.6), γ_4 0.638 (e_K/γ 0.0040), γ_5 0.723 (e_K/γ 0.0034) ($\gamma_2/\gamma_3/\gamma_4/\gamma_5 \approx 6/$ 100/10/3) spect, spect conv 0.080 (4.3%, with 0.283 γ , e/γ 1.83) γ - γ , x - γ coinc, scint spect
^{132}I		β^-	2.4 h 2.3 h	2.2, 0.9 abs 1.5 abs ~1.4 abs	γ_1 0.69, γ_2 1.41, γ_3 2.0 ($\gamma_1/\gamma_2/\gamma_3 \approx$ 37/4/1), $\gamma_4 \sim 0.8$ (very weak) scint spect, γ - γ coinc 0.6, 1.4 abs
^{133}I		β^-	20.5 h 22.4 h	1.3 (~91%), 0.4 (~9%) abs 1.4 (~94%), 0.5 (~6%) abs, β - γ coinc 1.4 abs	0.53 (94%), 0.85 (5%), 1.4 (1%) scint spect, γ - γ coinc 0.53 spect 0.55 abs

Isotope <i>Z</i>	<i>A</i>	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
					Particles	Gamma transitions
¹³⁴ La			β^-	52.5 m	1.6 (~70%), 2.8	>2.2 (weak) <i>D</i> - γ - <i>n</i>
				51 m	(~30%), hard β	reaction
				54 m	weak (abs)	
¹³⁶ I			β^-	6.68 h	1.5-1.75, 3.5-4.2 abs	>1 abs
				6.7 h	0.5 (35%), 1.0	1.8, 1.27 spect
					(40%), 1.4 (25%)	1.3 abs
					spect	1.6 abs
					1.4 abs	2.4 (1.1%) abs
¹³⁶ I			β^-		1.5 abs	
¹³⁶ I			β^-	86 s	6.5 abs	1.4, 2.9 scint spect, abs
¹³⁷ I			β^-, β^-n (~6% of disinte- grations)	22.0 s (<i>n</i>)	<i>n</i> (mean):	
				22.5 s (<i>n</i>)	0.56 abs paraffin	
				19.3 s genet	0.67 <i>p</i> recoil in cl ch	
¹³⁸ I			β^-	5.9 s		
¹³⁹ I			β^-	2.7 s		
¹³¹ Xe				40 m		
				70 m		
				~60 m		
¹³² Xe				19.5 h		
				20.0 h		
				19 h		
¹³³ Xe			β^+	2.1 h		
				1.7 h		
				~2 h		
¹³⁴ Xe	0.096					
¹³⁶ Xe			<i>EC</i> , no β^+	18 h		0.054 (<i>K/L</i> ~4.3),
				20 h		0.096, 0.106, 0.187
						(<i>K/L</i> ~4.6), 0.243
						spect conv; 0.460
						scint spect
¹³⁶ Xe	0.090					
¹³⁷ Xe			<i>IT</i> (?)	75 s		0.125, 0.175 spect
						conv
¹³⁷ Xe			<i>EC</i>	34 d		0.057, 0.145, 0.170,
				32 d		0.200, 0.365 spect
				25 d		conv, scint spect
¹³⁸ Xe	1.919					
^{129m} Xe			<i>IT</i>	8.0 d		0.196 (<i>K/L</i> + <i>M</i>
						2.1) spect conv
						0.040 (<i>K/L</i> + <i>M</i>
						4.3) spect conv
¹³⁹ Xe	26.44					
¹³⁰ Xe	4.08					

Isotope Z A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
				Particles	Gamma transitions
$^{131m}_54\text{Xe}$		<i>IT</i>	12.0 d		0.163 (<i>K/L</i> + <i>M</i> 1.9, <i>L/M</i> 3.4) spect conv 0.163 (<i>K/L</i> + <i>M</i> 1.7) spect conv 0.165 (<i>e</i> / <i>γ</i> ~20) abs conv, abs
Xe^{131m1}		<i>IT</i>	4.8×10^{-10} s delay coinc		0.080 scint spect
Xe^{131}	21.18				
Xe^{132}	26.89				
Xe^{133m}		<i>IT</i>	2.3 d 2.1 d		0.233 (<i>K/L</i> 2.9) spect conv 0.235 (<i>eK</i> / <i>γ</i> 4.2) spect conv, scint spect
Xe^{133}		β^-	5.270 d 5.3 d	0.345 spect 0.34 abs 0.35 abs	With Cs^{133m} : 0.081 (<i>K/L</i> 5.9) spect conv 0.08 <i>γ</i> (<i>eK</i> / <i>γ</i> 1.8, <i>K/L</i> + <i>M</i> 6.0) scint spect, β - <i>γ</i> delay coinc ~0.085 abs 0.08 cl ch
Xe^{134}	10.44				
Xe^{135m}		<i>IT</i>	15.6 m 15.3 m 13 m		0.52 spect 0.5 (<i>e</i> / <i>γ</i> ~0.2) abs, abs conv
Xe^{135}		β^-	9.13 h 9.2 h 9.1 h	0.905 spect 0.93 spect 0.95 abs 0.9 abs 1.0 abs	With Cs^{135m} : 0.250 (<i>eK</i> / <i>γ</i> 0.05) spect conv, β^- - conv coinc 0.248 (<i>K/L</i> 7.0) spect conv, scint spect, β - <i>γ</i> delay coinc 0.25 spect
Xe^{136}	8.87				
Xe^{137}		β^-	3.9 m 3.8 m 3.4 m	~4 abs	
Xe^{138}		β^-	17 m		
Xe^{139}		β^-	41 s		
Xe^{140}		β^-	16.0 s 9.8 s		
Xe^{141}		β^-	1.7 s 3 s		

Isotope Z A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
				Particles	Gamma transitions
^{142}Xe		β^-	1.0 s		
^{144}Xe		β^-	~ 1 s		
^{136}Cs		β^+	45 m	2.03 spect	
^{137}Cs		β^+	5.5 h	1.2 spect, abs	
^{138}Cs		β^+ , EC	3.8 m 3.1 m	3.0 abs	
^{139}Cs		EC, no β^+	31 h	Conv: ~ 0.3 abs	~ 0.5 abs
^{140}Cs		β^+ , EC, β^- (β^+/β^- 27.5)	30 m ~ 30 m	β^+ : 1.97 spect β^- : 0.442 spect	No γ Xe K-x
^{141}Cs		EC, no β^+	9.6 d 10.2 d 10.0 d		No γ Xe K-x ~ 0.1 abs conv, abs
^{132}Cs		EC	7.1 d		0.668 scint spect 0.62 abs, abs conv
$^{133\text{m}}\text{Cs}$		IT	6.0×10^{-9} s delay coine		See ^{133}Xe ~ 0.061 (e_K/γ 1.8, K/L + M 6.0) scint spect, β - γ delay coine
^{133}Cs	100				
$^{124\text{m}}\text{Cs}$		IT	3.2 h 3 h		0.128 (K/L/M $\approx$ 64.3/100/18.6) spect conv 0.128 (e_K/γ 2.2) scint spect 0.128 ($L_{II}/L_{III} \sim 1$) spect conv
^{134}Cs		β^- No EC (lim 4%) No EC (lim 5%) No β^+ (lim 0.009%)	2.3 y 1.7 y	0.648 (75%), 0.09 (25%) spect 0.65 spect 0.66 ($\sim 72\%$), 0.09 ($\sim 28\%$) spect 0.676, 0.640, ~ 0.08 ($\sim 24\%$) spect 0.60, 0.09 abs, β - γ coine abs	0.561 (e_K/γ 0.005), 0.567 (e_K/γ 0.007), 0.601 (e/γ 0.005, K/L 6.0), 0.794 (e/γ 0.002, K/L 6.0), 1.037 (weak, K/L 4.5), 1.164 (weak, K/L 6.2), 1.365 (weak, K/L 6.1) spect conv 0.560 (e/γ 0.008), 0.602 (e/γ 0.0053), 0.799 (e/γ 0.0025), 1.037 (weak), 1.170 (weak), 1.363 (with 0.09 β^- , e/γ 0.00062) spect conv, β - γ coine 0.040 crit abs

Isotope Z A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
				Particles	Gamma transitions
$^{134}_{55}\text{Cs}$ (Cont.)					γ_1 0.570, γ_2 0.601, γ_3 0.793, γ_4 1.024, γ_5 1.11, γ_6 1.35 ($\gamma_1/\gamma_2/\gamma_3/\gamma_4 \approx$ 0.35/0.94/1.0/ 0.017) spect γ_1 0.57, γ_2 0.60, γ_3 0.79 ($\gamma_1/\gamma_2/\gamma_3$ $\approx$ 0.26/1.0/1.0) spect No 1.96 γ (lim $10^{-4}\%$) Be- γ -n reaction
$\text{Cs}^{136\text{m}}$		IT	2.8×10^{-10} s delay coin		0.248 (K/L 7.0) spect conv, scint spect, β - γ delay coin
Cs^{135}		β^-	3.0×10^6 y sp act 2.1×10^6 y yield	0.21 abs ~ 0.19 abs	No γ
Cs^{136}		β^-	13.7 d	0.35 abs 0.28 β - γ coin abs	~ 0.9 abs 1.2 β - γ coin abs Two γ 's
Cs^{137}		β^-	33 y 30 y	β_1 0.523 spect β_1 0.51 (92%), β_2 1.17 (8%) spect β_1 0.521, $\beta_2 \sim 1.2$ spect β_1 0.518, $\beta_2 \sim 1.18$ spect Others	With $\text{Ba}^{137\text{m}}$: 0.6616 cryst spect See $\text{Ba}^{137\text{m}}$: Others
Cs^{138}		β^-	32.9 m 33 m 32 m	3.40 (coin with 1.4 γ), ~ 2.9 , ~ 2.0 spect, β - γ coin abs 2.68 spect 2.65 abs	γ_1 0.463, γ_2 0.98, γ_3 1.44 (coin with γ_1 and γ_2) spect conv, scint spect, β - γ , γ - γ coin 1.2 abs
Cs^{139}		β^-	9.5 m 10 m 7 m		
Cs^{140}		β^-	66 s		
Cs^{141}		$[\beta^-]$	Short		
Cs^{142}		β^-	~ 1 m		
Cs^{143}		$[\beta^-]$	Short		
Cs^{144}		$[\beta^-]$	Short		
$^{127}_{86}\text{Ba}$			~ 12 m		
Ba^{128}		EC	2.4 d		

Isotope Z	A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
					Particles	Gamma transitions
⁵⁴ Ba ¹²⁹			β^+	2.0 h 1.8 h	Hard β^+	
Ba ¹³⁰	0.101					
Ba ¹³¹			EC No β^+	12.0 d 11.7 d		γ_1 0.122 (K/L 6.0), γ_3 0.214 (K/L 2.8, $e_K/\gamma \sim 0.18$), γ_4 0.241, γ_5 0.370 ($e_K/\gamma \sim 0.01$), γ_6 0.494 (K/L 2.5, $e_K/\gamma \sim 0.005$), ~ 0.043 (?), ~ 0.065 (?), ~ 0.108 (?) (γ_3 / $\gamma_4/\gamma_5/\gamma_6 \approx 10$ / 4/7/100) spect, spect conv γ_1 0.122, γ_2 0.196, γ_3 0.213, γ_4 0.241, γ_5 0.371, γ_6 0.497 ($\gamma_1/\gamma_2/\gamma_3/\gamma_4/\gamma_5$ / $\gamma_6 \approx 1.3/4.4$ / 5.3/1.4/12/100) spect 0.122 (e/γ 0.8, K/L ~ 3.5), 0.206 (e/γ 0.15), 0.372 (e/γ 0.03), 0.494 (e/γ 0.01) spect, spect conv, γ - γ , conv- conv coinc ~ 0.5 (coinc with ~ 0.2 γ) γ - γ coinc abs γ_7 1.2 abs ($\gamma_6/\gamma_7 \sim 4$)
Ba ¹³²	0.097					
Ba ^{133m}			IT	38.8 h 38.9 h		0.276 ($e_K/\gamma \sim 3$) spect conv, scint spect 0.0117 ($e/\gamma \sim 130$) γ - γ coinc, ion ch 0.276 (K/L 3.2) spect conv
Ba ¹³³			EC	~ 9.5 y		0.320 (e/γ 0.02), 0.085 (e/γ 0.3, K/L ~ 10) abs, abs conv, cl ch 0.36 abs, abs conv
Ba ¹³⁴	2.42					
Ba ^{135m}			IT	28.7 h		0.269 ($e_K/\gamma \sim 3.5$, K/L ~ 2) spect conv, scint spect 0.267 spect conv, spect, scint spect

Isotope Z	A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
					Particles	Gamma transitions
$_{56}\text{Ba}^{135}$		6.59				
Ba^{136}		7.81				
$\text{Ba}^{137\text{m}}$			<i>IT</i>	2.60 m 2.63 m 2.5 m		0.6616 cryst spect γ_1 0.661 ($K/L + M$ 4.64) spect conv 0.661 ($K/L/M + N$ $\approx 5.5/1.0/0.27$) spect conv γ_1 (eK/γ 0.097) spect, spect conv γ_1 (eK/γ 0.095) scint spect 0.662 ($K/L + M$ 4.57) spect conv 0.663 (eK/γ 0.13, K/L 4.8) spect conv γ_1 (eK/γ 0.08, K/L 5.0) spect conv 0.663 (e/γ 0.14) spect conv, x -conv coine 0.669 spect conv
Ba^{137}		11.32				
Ba^{138}		71.66				
Ba^{139}			β^-	85.0 m 84 m 86 m	2.27 spect 2.3 abs	0.163 (26%, e/γ 0.20, $K/L \approx 6$), 1.05 (0.6%) spect conv, abs, coine ~ 0.163 (eK/γ 0.28) scint spect
Ba^{140}			β^-	12.80 d	1.022 (60%), 0.480 spect 1.05 spect 0.99, 0.47 spect 1.0 ($\sim 75\%$), ~ 0.4 ($\sim 25\%$) abs	0.0296, 0.132, 0.162, 0.304, 0.537 spect conv 0.160, 0.310, 0.535 spect 0.03, 0.16, 0.31, 0.54 spect, spect conv, scint spect, γ - γ coine
Ba^{141}			β^-	18 m	2.8 abs	γ
Ba^{142}			β^-	6 m		
Ba^{143}			β^-	<0.5 m		
Ba^{144}			β^-	Short		
$_{57}\text{La}^{131}$			β^+	58 m	1.6 abs	
La^{132}			β^+	4.5 h	3.5 abs	1.0 abs
La^{133}			<i>EC</i> , β^+ (weak)	4.0 h	~ 1.2 abs, spect Conv: 0.26 spect conv	0.8 abs
La^{134}			$\beta^+ \sim 44\%$, <i>EC</i> $\sim 56\%$	6.5 m	2.7 abs, spect	No γ

Isotope Z A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
				Particles	Gamma transitions
^{135}La		EC	19 h 19.5 h		0.76 (weak) abs 0.88 abs
^{136}La		EC $\sim 67\%$, β^+ $\sim 33\%$	9.5 m 9.0 m 10 m	2.1 spect, abs 1.8 abs	
^{137}La			>400 y yield >30 y yield		
^{138}La	0.089	EC $\beta^- \sim 6\%$	$\sim 2.0 \times 10^{11}$ y sp act $\sim 7 \times 10^{10}$ y sp act	1.0 abs	γ_1 1.39, γ_2 0.81, γ_3 0.54 ($\gamma_1/\gamma_2/\gamma_3 \approx$ 1/0.65/0.3) scint spect 1.0, 0.54 scint spect
^{139}La	99.911				
^{140}La		β^-	40.0 h 40.4 h 39.5 h	1.32 ($\sim 70\%$), 1.67 ($\sim 20\%$), 2.26 ($\sim 10\%$), others <1.3 (very weak) spect 0.90, 1.40, 2.12 spect 1.45, 2.2 spect Others	γ_2 0.3286, γ_3 0.4867, γ_4 0.8151, γ_5 1.596 spect γ_1 0.093, γ_2 0.335, γ_3 0.490, γ_4 0.820, γ_5 1.600, γ_6 2.50, γ_7 3.0 (weak) spect, spect conv, γ - γ coinc γ_1 0.093, γ_2 0.335, γ_3 0.49, γ_4 0.82, γ_5 1.62, γ_6 2.55 ($\gamma_1/\gamma_2/\gamma_3/\gamma_4/\gamma_5/$ $\gamma_6 \approx <1/3/22/$ 18/56/3) scint spect 0.069, 0.110, 0.131, 0.173, 0.241, 0.265, 0.329, 0.431, 0.486, 0.752, 0.816, 0.926, 1.597, 1.904 spect conv γ_2 0.335, γ_3 0.49, γ_4 0.87, γ_5 1.65, γ_6 2.3 ($\gamma_2/\gamma_3/\gamma_4/$ $\gamma_5/\gamma_6 \approx 2/5/10/$ 77/6) spect γ_2 0.335, γ_3 0.49, γ_4 0.83, γ_5 1.63, $\gamma_6 \sim 2.3$ ($\gamma_2/\gamma_3/$ $\gamma_4/\gamma_5/\gamma_6 \approx 1/10/$ 20/100/5) spect γ_5 (coinc with 2.3 β^- and γ_4) scint spect, β - γ , γ - γ coinc 2.55 ($\sim 4\%$), 2.9 ($\sim 0.1\%$) D- γ -p ion ch 2.49 ($\sim 3\%$) D- γ -n, Be- γ -n reactions Others

Isotope Z	A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
					Particles	Gamma transitions
$_{57}\text{La}^{141}$			β^-	3.7 h 3.5 h	2.43 ($\sim 95\%$), 0.9 ($\sim 5\%$) spect	1.3-1.6 (?), weak scint spect, β - γ coinc
La^{142}			β^-	74 m 77 m		γ
La^{143}			β^-	~ 19 m genet ~ 15 m		
La^{144}			$[\beta^-]$	Short		
$_{58}\text{Ce}^{133}$			EC, β^+	6.30 h	1.3 spect, abs	1.8 abs
Ce^{134}			EC	72.0 h		K-x, no γ
Ce^{135}			EC, $\beta^+ \leq 1\%$	22 h	0.81 spect	
Ce^{136}	0.193					
Ce^{137}			EC, no β^+	36 h		0.257 (K/L ~ 4) spect conv 0.253 (K/L ~ 10) spect conv
Ce^{138}	0.250					
Ce^{139}			EC	140 d		0.166 (K/L ~ 10), 0.275 spect conv 0.166 (K/L ≥ 4) spect conv ~ 0.8 abs
Ce^{140}	88.48					
Ce^{141}			β^-	33.1 d 32.5 d 30.6 d	0.581 (33%), 0.442 (67%) spect 0.58 (29%), 0.44 (71%) spect 0.56 (30%), 0.41 (70%) spect 0.56 spect, β - γ coinc abs Others	0.145 (e_K/γ 0.25, K/L 5.5) spect conv, spect ~ 0.14 (e_K/γ 0.46) scint spect 0.142 (e_K/γ 0.48) scint spect 0.145 (K/L ~ 7) spect conv 0.144 ($s/\gamma \sim 0.33$, K/L 6.5) spect conv 0.146 spect conv 0.141 (coinc with β^-) spect conv, β - conv coinc 0.14 (coinc with β^-) scint spect, β - γ coinc Others
Ce^{142}	11.07					
Ce^{143}			β^-	33 h 34 h 36 h	β_1 1.39, β_2 1.09, β_3 0.71 ($\beta_1/\beta_2/\beta_3 \approx 1.0/1.3/$ 1.0) spect β_1 1.37, β_2 1.09, β_3 0.37 (?) (β_2/β_1 1.4) spect	0.035, 0.126, ~ 0.160 , 0.289, 0.356, 0.660, 0.720 spect, spect conv, scint spect

Isotope Z A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
				Particles	Gamma transitions
¹⁴³ Ce (Cont.)					γ_1 0.057 ($K/L \sim 1$), γ_2 0.283 ($K/L \sim 6$), γ_3 0.649, γ_4 0.705 ($\gamma_2/\gamma_3/\gamma_4 \approx 4.5/1/1$) spect, spect conv, β - γ coinc 0.0575 ($K/L < 1$), 0.291 ($K/L \sim 10$), 0.348 spect conv Others
¹⁴⁴ Ce		β^-	282 d 275 d 290 d	0.300 (70%), 0.170 (30%, coinc with 0.134 γ , with 0.080 γ) spect, β - γ coinc 0.304 (70%) spect 0.17 (coinc with 0.134 γ) β - γ coinc Others	0.0337, 0.054, 0.0807 (e/γ large, $K/L \sim 4$), 0.100, 0.134 ($K/L \sim 9$) spect conv, spect 0.034, 0.041, 0.047, 0.054 ($K/L < 1$), 0.081 ($K/L \sim 5$), 0.095, 0.101, 0.135 ($K/L \sim 10$) spect conv 0.0547, 0.079 (K/L 6.3), 0.134 (K/L 8.3), 0.231 (K/L 1.7) spect conv ~ 0.132 (K/L 5.3) spect conv 0.695, 1.50, 2.18 spect Others
¹⁴⁶ Ce			~ 1.8 h		
¹⁴⁶ Ce		β^-	14.6 m 11 m	~ 0.9 abs	Soft γ
¹³⁷ Pr		β^+	1.4 h	1.8	
¹³⁸ Pr		EC $\sim 90\%$, β^+ $\sim 10\%$	2.0 h	1.4 abs, spect	0.2, ~ 0.5 , 1.3 abs
¹³⁹ Pr		EC $\sim 94\%$, β^+ $\sim 6\%$	4.2 h 4.5 h	1.0 abs spect	1.0 abs
¹⁴⁰ Pr		β^+ 58%, EC 42%	3.4 m 3.5 m	2.23 spect	No γ
¹⁴¹ Pr	100				
¹⁴² Pr		β^- No β^+ or EC (lim 0.5%)	19.2 h 19.1 h 19.3 h	2.15 ($\sim 96\%$), 0.64 ($\sim 4\%$) spect 2.23, 0.66 spect 2.14 spect 2.23 spect 2.22, 0.22 abs, β - γ coinc abs 2.5, ~ 0.4 abs, β - γ coinc abs	γ_1 0.135, γ_2 1.59 ($\gamma_1/\gamma_2 < 0.2$) spect, spect conv 1.58 spect 0.134, 0.329, 0.490, 0.624 spect conv; 2.1 abs 1.5 coinc abs sec
¹⁴³ Pr		β^-	13.7 d 13.8 d 13.5 d	0.932 spect 0.922 spect 0.920 spect 0.92 spect 0.84 abs	No γ

Isotope Z	A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
					Particles	Gamma transitions
⁸⁹ Pr ¹⁴⁴			β^-	17.5 m	2.97 (>99%), other β 's	γ_1 0.0603, γ_2 0.696,
				17 m	(<1%) spect	γ_3 1.5, γ_4 2.19 (γ_2 ,
					2.32 (<1%, coine with	γ_3 , γ_4 weak) scint
					0.69 γ), 0.81 (<1%)	spect, spect, spect
					spect, β - γ coine	conv
					2.95 (~95%), 0.87	γ_2 0.695, γ_3 1.48, γ_4
Pr ¹⁴⁶			β^-	4.5 h	(~5%) spect	2.19 ($\gamma_2/\gamma_3/\gamma_4 \approx$
					2.97 (~90%), 2.3	1/0.4/1.1) spect,
					(~5%), 0.86 (~5%)	β - γ , γ - γ coine
					spect	γ_1 0.061 ($K/L < 1$)
					2.99 spect	spect conv
Pr ¹⁴⁶			β^-	24.0 m	3.8 abs	0.490, 0.78 scint
				24.6 m	~3 abs	spect
				25 m		1.4 abs
⁶⁰ Nd ¹³⁸			β^+	22 m	~2.4 abs	
Nd ¹³⁹			EC ~90%, β^+ ~10%	5.50 h	3.1 abs, spect	1.3 abs
Nd ¹⁴⁰			EC	3.3 d		Pr K-x
Nd ¹⁴¹			EC ~98%, β^+ ~2%	2.42 h	0.7 abs	1.2 (weak) abs
				2.5 h	0.8 abs	
Nd ¹⁴²	27.13					
Nd ¹⁴³	12.20					
Nd ¹⁴⁴	23.87					
Nd ¹⁴⁶	8.30					
Nd ¹⁴⁶	17.18					
Nd ¹⁴⁷			β^-	11.3 d	0.83 (~60%), 0.60	0.0918 ($K/L_1 \approx 6.4$)
					(~15%), 0.38	spect conv
					(~25%) spect	γ_1 0.0918 ($e/\gamma \sim 0.9$,
					0.78 (~65%), 0.35	$K/L + M$ 6.5) γ_2
					(~32%) spect	0.309, γ_3 0.391,
					0.83, 0.60, 0.38 spect	γ_4 0.520 ($\gamma_1/\gamma_2/$
						$\gamma_3/\gamma_4 \approx 66/1/2/$
						32) spect
						γ_1 ($K/L/M \approx 7.55/$
						1/0.096) spec conv
						0.0912 (K/L 4.9),
						0.121, 0.197,
						0.231, 0.260,
						0.273, 0.301,
						0.318, 0.398,
						0.441, 0.532 (K/L
						~6) (all weak
						except 0.09 γ)
						spect conv, γ - γ ,
						β - γ coine
						0.0915 (coine with
						0.83 β^-), 0.320,
						0.534 spect conv,
						β -conv coine
						Others

Isotope Z A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
				Particles	Gamma transitions
⁶⁰ Nd ¹⁴⁸	5.72				
Nd ¹⁴⁹		β^-	2.0 h 1.8 h 1.7 h	1.5, 1.1, 0.95 spect 1.5 abs 1.6 abs	0.030, 0.097 (K/L 0.9), 0.112, 0.114 (K/L $\sim$ 5), 0.124, 0.188, 0.198, 0.211 (K/L $\sim$ 7), 0.226, 0.240, 0.266 (K/L $\sim$ 10), 0.424, 0.538, 0.650 spect, spect conv, scint spect, coinc Others
Nd ¹⁵⁰	5.60	β^- (?)	$>2 \times 10^{15}$ y sp act $>10^{13}$ y sp act		
Nd ¹⁵¹		β^-	15 m 12 m	1.93 spect	0.085, 0.110, 0.117 (K/L 4), 0.421, 0.73, 1.14 spect conv, scint spect, β - γ , γ - γ coinc Pm K-x
⁶¹ Pm ¹⁴¹		β^+	20 m	2.4-2.8 spect	
Pm ^{142,143}		EC	250-280 d 285 d		0.95 abs
Pm ^{143,144}		EC	200-400 d 300-350 d		0.65, 0.44, 0.17 scint spect
Pm ¹⁴⁵		β^+	14-18 d	0.45	
Pm ¹⁴⁶		EC	$\sim$ 30 y yield		Nd K, L-x
Pm ¹⁴⁶		β^- (?)	$\sim$ 1 y 1-2 y	0.7 abs 0.75	
Pm ¹⁴⁷		β^-	2.6 y 2.3 y yield	0.223 spect 0.227 spect 0.229 spect	No γ
Pm ¹⁴⁸		β^-	5.3 d	$\sim$ 2.5 abs 2 abs	$\sim$ 0.8 abs
Pm ¹⁴⁸		β^-	42 d 43 d 48 d	2.4 (weak), 0.6 spect 2.7 (weak), 0.7 abs 1.7, 0.6 abs	0.9 abs 1.0 abs 0.5 abs
Pm ¹⁴⁹		β^-	54 h 55 h 50 h 47.5 h 47 h	1.05 spect	0.285 (coinc with β^- , K/L 8), $\sim$ 1.3 (weak) spect conv, abs, β - γ coinc No γ $\sim$ 0.2 (coinc with β^-) β - γ coinc abs
Pm ¹⁵⁰		β^-	2.7 h	2.01 ($\sim$ 70%), 3.00 ($\sim$ 30%) spect 2.4 abs	1.4, 0.3 abs

Isotope Z A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
				Particles	Gamma transitions
$^{61}\text{Pm}^{161}$		β^-	27.5 h	1.1 abs	0.065 (K/L 0.3), 0.066 (K/L 0.3), 0.070 (K/L 0.3), 0.100 (K/L ~ 5), 0.116, 0.144 (K/L ~ 9), 0.163 (K/L ~ 7), 0.168 (K/L ~ 3), 0.177 (K/L ~ 9), 0.208 (K/L ~ 4), 0.232, 0.240, 0.275 (K/L >10), 0.340 (K/L ~ 9), 0.715 spect conv, scint spect
Pm		β^-	12.5 h		
$^{62}\text{Sm}^{143}$			8 m		
Sm^{144}	3.16				
Sm^{146}		EC	~ 410 d >150 d >72 d		0.061 (K/L 1.0) spect conv
Sm^{147}	15.07	α	$t_{1/2}$ corrected for abun- dance of Sm^{147} 1.4×10^{11} y sp act 1.5×10^{11} y sp act 1×10^{11} y sp act	2.18 ion ch 2.14 range emuls 2.1 range emuls 2.0 el ch Others	
Sm^{148}	11.27				
Sm^{149}	13.84				
Sm^{150}	7.47				
Sm^{151}		β^-	73 y ~ 120 y yield	0.076 spect 0.079 spect 0.074 spect No conv	0.019 (coinc with β^-) ion ch, β - γ coinc 0.021 ion ch 0.021 scint spect No γ Others
Sm^{152}	26.63				
Sm^{153}		β^-	47 h	0.80, 0.70 (coinc with 0.101 γ) scint spect, β - γ coinc 0.68 ($\sim 67\%$), 0.80 ($\sim 33\%$) spect 0.70 spect 0.82 spect 0.78 abs 0.73 abs	γ_1 0.070 (eK/γ 3.1), γ_2 0.104 (eK/γ 1.2, coinc with γ_3 or γ_4), γ_3 0.530, γ_4 ~ 0.60 ($\gamma_1/\gamma_2/\gamma_3$ / $\gamma_4 \approx 100/425$ / 1.0/0.3) scint spect, γ - γ coinc 0.069 (eK/γ 3.8), 0.103 (eK/γ 1.2, $K/L + M$ 3.5) scint spect

Isotope Z A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
				Particles	Gamma transitions
⁶² Sm ¹⁵³ (Cont.)					0.070, 0.103 (coinc with 0.07 γ , eK/γ 0.65, $K/L \sim 6$), 0.530 (weak) scint spect, spect conv, conv-conv coinc 0.070 (weak, e/γ >10), 0.101 (e/γ ~ 3), no higher γ , scint spect, γ - γ coinc 0.070 (K/L 3.5, $L_1/L_{11}/L_{111} \approx 26/1.3/1.3$), 0.104 (K/L 6.5, $L_1/L_{11} + L_{111} \approx 43/2.3$) spect conv 0.070 (coinc with 0.103 γ , K/L 0.29), 0.103 (K/L 3.5), 0.582 (weak) spect conv, γ - γ coinc 0.069, 0.103 spect conv 0.102 ($e/\gamma > 2.5$) spect conv, β - γ , β -conv coinc
Sm ¹⁵⁴	22.53				
Sm ¹⁵⁵		β^-	23.5 m 25 m 21 m	1.8 (coinc with both γ 's) abs, β - γ coinc 1.9 abs 1.8 abs	γ_1 1.05 (K/L 3.6), γ_2 0.246 (coinc with γ_1 , $K/L \sim 8$) ($\gamma_1/\gamma_2 \sim 1$) spect, spect conv, γ - γ coinc
Sm ¹⁵⁶		β^-	~ 10 h	0.9 abs	
⁶³ Eu ¹⁴⁴		β^+	18 m	2.4 spect	
Eu ¹⁴⁵		EC	5 d	Conv: 0.2 abs	
Eu ¹⁴⁶		EC	38 h	Conv: 0.4 abs	
Eu ¹⁴⁷		EC 99+%, α $\sim 10^{-3}\%$, no β^+	24 d	α : 2.88 ion ch Conv: 0.2 abs	
Eu ¹⁴⁸		EC, no β^+	59 d 54 d 50 d 53 d	Conv: 0.4 abs	0.57 scint spect 1.0, 0.4 abs 0.7 abs
Eu ¹⁴⁹			~ 120 d		~ 0.4 scint spect
Eu ¹⁵⁰		β^-	15.0 h 15 h 13.1 h	1.8 spect 0.8, other β 's	
Eu ¹⁵¹	47.77				

Isotope Z A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
				Particles	Gamma transitions
^{152}Eu		EC, β^-	13 y 5.3 y yield	1.58, others spect 1.58, 0.75 spect 1.7 (~20%), 0.9 (~80%) abs	0.122 (conv in Sm), 0.123 (conv in Gd), 0.244, 0.344, 0.720, 0.964, 1.086 spect conv 0.121, 0.244, 0.344 spect conv 0.121 (coinc with 0.244 γ), 0.123 (coinc with 0.344 γ), 0.244, 0.344 (not coinc with 0.244 γ) conv- conv coinc spect Others
^{152}Eu		β^-, EC	9.2 h 9.3 h	1.880, 0.55 (?) (weak) spect 1.88 spect	γ_1 0.122 (conv in Sm, K/L ~4), γ_2 0.344 (conv in Gd, K/L ~10) spect conv 0.122, 0.336 (coinc with β) pair spect, β - γ coinc 0.120 (coinc with 0.9 or 0.8 γ), 0.94, 0.82 (not coinc with 0.9 γ) spect, spect conv, β -conv, γ - γ coinc Others
$^{153\text{m}}\text{Eu}$		IT	3.0×10^{-9} s delay coinc		0.069 (eK/γ 3.8), 0.103 (eK/γ 1.2, K/L + M 3.5) scint spect
^{153}Eu	52.23				
^{154}Eu		β^-	16 y 5.4 y yield	1.9 (~10%), 0.7 (~40%), 0.3 (~50%) abs ~0.7 (coinc with hard γ), 0.3 abs, β - γ coinc abs With ^{154}Eu and ^{156}Eu : 1.88, 0.90, 0.59, 0.25, 0.14 spect, β - γ coinc abs Others	0.336, 0.778, 1.116 spect conv With ^{154}Eu and ^{156}Eu : 0.085, 0.101, 0.725, 1.005, 1.288 spect 0.122 spect conv Others
^{156}Eu		β^-	1.7 y yield 2.0 y	0.154 (80%), 0.243 (20%) spect, β - γ , γ - γ coinc See ^{154}Eu β 's	0.060 (weak), 0.087 (K/L ~8), 0.106 (K/L ~8), 0.132 (weak) spect conv 0.085, 0.099 crit abs Pb, Pt 0.084 crit abs Tl, Hg 0.015 ion ch, β - γ coinc See ^{154}Eu γ 's

Isotope Z A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
				Particles	Gamma transitions
$^{155}_{63}\text{Eu}$		β^-	15.4 d	~ 0.5 (60%), 2.4 (40%) abs	2.0 abs
$^{157}_{63}\text{Eu}$		β^-	15.4 h	~ 1.0 ($\sim 75\%$), ~ 1.7 ($\sim 25\%$) abs	0.6, 0.2 abs
$^{158}_{63}\text{Eu}$		β^-	60 m	2.6 abs	γ
$^{159}_{63}\text{Eu}$			20 m		
$^{148}_{64}\text{Gd}$		α , EC (?)	>35 y	α : 3.16 ion ch	
$^{149}_{64}\text{Gd}$		EC 99+%, $\alpha \sim 10^{-3}\%$	9 d	α : 3.0 ion ch Conv: 0.35 abs	
$^{150}_{64}\text{Gd}$		α	Long	α : 2.7 ion ch, range emuls	
$^{151}_{64}\text{Gd}$		EC, no β^+	150 d		0.265 (e/ γ large) abs
$^{152}_{64}\text{Gd}$	0.20				
$^{153}_{64}\text{Gd}$		EC, no β^+	236 d 225 d		0.104 (K/L 5.2) spect conv 0.100 spect conv, abs 0.106 (e/ γ >0.9) abs
$^{154}_{64}\text{Gd}$	2.15				
$^{155}_{64}\text{Gd}$	14.73				
$^{156}_{64}\text{Gd}$	20.47				
$^{157}_{64}\text{Gd}$	15.68				
$^{158}_{64}\text{Gd}$	24.87				
$^{159}_{64}\text{Gd}$		β^-	18.0 h 17.9 h ~ 24 h	0.9 abs	0.055, 0.38 abs ~ 0.3 abs
$^{160}_{64}\text{Gd}$	21.90				
$^{161}_{64}\text{Gd}$		β^-	3.6 m 3.5 m 3.3 m 4.5 m	1.5 abs ~ 2	0.37 abs ~ 0.07
$^{149}_{65}\text{Tb}$		α , EC (?)	4.1 h	α : 3.955 spect 3.95 ion ch	
$^{161}_{65}\text{Tb}$		α , EC (?)	19 h	3.44 ion ch	
$^{163}_{65}\text{Tb}$		EC	5.1 d		1.2, 0.2 abs
$^{164}_{65}\text{Tb}$		EC 99+%, $\beta^+ \sim 0.5\%$	17.2 h	2.6 spect	1.3 abs
$^{166}_{65}\text{Tb}$		EC	190 d		1.4 abs
$^{168}_{65}\text{Tb}$		EC >75%, $\beta^+ < 25\%$	5.0 h	~ 1.3 abs	
$^{167}_{65}\text{Tb}$		EC	4.7 d		1.4 abs
$^{169}_{65}\text{Tb}$	100				

Isotope Z A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
				Particles	Gamma transitions
⁶⁶ Tb ¹⁶⁰		β^-	73.5 d	0.860 (43%), 0.521	0.962, 0.876, 0.410,
		No β^+	71 d	(41%), 0.396 (16%)	0.391, 0.375,
			76 d	spect	0.298, 0.282,
				$\sim 0.90 \beta^-$ (coinc with 0.085 γ) scint spect, β - γ coinc	0.215, 0.196, 0.176, 0.093, 0.087 spect, spect conv 0.970, 0.886, 0.300, 0.200, 0.085 spect conv With Dy ^{160m} : 0.085 (eK/γ 1.7) scint spect Others
Tb ¹⁶¹		β^-	6.75 d	0.5 abs	0.049 (L/M 3.7)
			6.8 d		spect conv
			7.2 d		0.026 ion ch
			7 d		0.05 abs
Tb ^{162, 163}			14 m		
⁶⁶ Dy ^{<167}		α	7 m	4.21 ion ch	
Dy ¹⁶³		α	19 m	4.06 ion ch	
Dy ¹⁶⁴		α	2.3 h	3.61 ion ch	
Dy ¹⁶⁶	0.0524				
Dy ¹⁶⁸	0.0902				
Dy ¹⁶⁹		EC	134 d 140 d		Tb K, L-x
Dy ^{160m}		IT	1.8×10^{-9} s delay coinc		0.085 (eK/γ 1.7) scint spect
Dy ¹⁶⁰	2.294				
Dy ¹⁶¹	18.88				
Dy ¹⁶²	25.53				
Dy ¹⁶³	24.97				
Dy ¹⁶⁴	28.18				
Dy ^{165m}		IT	1.25 m		0.109 (K/L 0.08) spect conv 0.102 scint spect 0.102 spect conv
Dy ¹⁶⁶		β^-	139.2 m 140 m 145 m	1.25, 0.88, 0.42 spect	0.0951 (K/L _I 6.4) spect conv 0.0878 spect conv 0.091, 0.36, 0.76 spect, spect conv
Dy ¹⁶⁸		β^-	82 h 81 h	0.2 abs 0.4 abs	
⁶⁷ Ho		α	4 m	α : 4.2 ion ch	

Isotope Z A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
				Particles	Gamma transitions
^{160}Ho		EC 99+%, β^+ ~0.5%	22.5 m	~1.3 abs Conv: 0.2 abs	~1.2 abs
^{161}Ho		EC	4.6 h	Conv: 0.1 abs	1.1 abs
^{162}Ho		EC ~85%, β^- ~15%	65.0 d	β^- : 0.8 spect, abs Conv: 0.1 abs	~1 abs
^{163}Ho		EC	5.20 d	Conv: 0.4 abs	~0.5, 1.4 abs
^{164}Ho		β^-	34.0 m 41.5 m	0.95 spect	No γ
^{165}Ho	100				
^{166}Ho		β^-	27.3 h	1.84 (~89%), 0.55 (~11%) spect 1.88 spect 1.84 (86%), 0.66 (14%) spect 1.85 spect 1.90 cl ch, abs	0.080 (eK/γ 1.9, $K/L + M$ 0.25), 1.38 (coinc with 0.08 γ) scint spect, γ - γ coinc 0.081 (K/L 0.07, $L_{II}/L_{III}/L_{III} \approx$ <0.1/0.72/1.00) spect conv 0.080 (eL/γ ~0.4, K/L <1), 1.36 (weak) spect, spect conv, β -conv coinc 0.081 (eK/γ 1.9) scint spect 0.081, 0.9 spect conv, abs 0.081, ~1.5 (weak) spect conv, β - γ coinc 0.080, 1.2 (weak) spect conv, abs
^{166}Ho		β^-	>30 y	1.1 (~8%), 0.28 (~46%), 0.18 (~46%) abs	0.212 (coinc with 1.1 β^-), 0.280 (coinc with 0.73 and 0.83 γ), 0.725, 0.830, 0.095 (very weak) scint spect, γ - γ , β - γ coinc
$^{167,169}\text{Ho}$			96 m		
$^{160,161}\text{Er}$		β^+ (?)	~17 h		
^{162}Er	0.136				
^{163}Er		β^+ (?)	~65 h		
^{164}Er	1.56				
^{165}Er		EC	10.0 h 9.9 h 11.2 h	Conv: ~0.2, 1.1 (weak) abs	1.1 abs No γ
^{166m}Er		IT	1.7×10^{-9} s delay coinc		0.081 (eK/γ 1.9) scint spect 0.081 (K/L 0.07) spect conv

Isotope Z	A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
					Particles	Gamma transitions
⁶⁸ Er ¹⁶⁶		33.41				
Er ¹⁶⁷		22.94				
Er ¹⁶⁸		27.07				
Er ¹⁶⁹			β^-	9.4 d 9 d	0.33 spect 0.33 scint spect	No γ
Er ¹⁷⁰		14.88				
Er ¹⁷¹			β^-	7.5 h	1.49 (~6%), 1.05 (~72%), 0.67 (~22%) spect, β - γ coinc	0.113 (K/L ~10), 0.118 (K/L ~0.5), 0.126 (K/L ~2), 0.176, 0.295 (K/L ~10), 0.308 (K/L ~10), 0.420, no 0.8 γ spect conv 0.113 (e/γ large, coinc with 1.05 β and 0.31 γ), 0.31 (coinc with 1.05 β), 0.81 spect, spect conv, β - γ coinc ~0.1 (e/γ 1.3) β - γ delay coinc
Er ^m			IT	2.5 s		0.210 (eK/γ 0.55) scint spect
⁶⁹ Tm ¹⁶⁶			EC 99+%, β^+ ~0.5%	7.7 h	β^+ : 2.1 spect Conv: 0.24, ~1 spect, abs	1.7 abs
Tm ¹⁶⁷			EC, no β^+	9.6 d	Conv: 0.21 abs	0.22, 0.95 abs
Tm ¹⁶⁸			EC, β^- (?) ~2%	85 d	Conv: 0.16, 0.5 abs	0.21, 0.85 abs
Tm ^{169m}			IT	All delay coinc: 0.658 $\times$ 10 ⁻⁶ s 0.67 $\times$ 10 ⁻⁶ s 0.7 $\times$ 10 ⁻⁶ s 0.60 $\times$ 10 ⁻⁶ s		See γ 's of Yb ¹⁶⁹
Tm ¹⁶⁹		100				
Tm ¹⁷⁰			β^- No EC (lim 0.3%), no β^+ (lim 0.01%)	129 d 120 d 127 d	0.968 (76%), 0.884 (24%) spect 0.970 (~90%), 0.886 (~10%) spect 0.970, 0.88 spect, β - γ coinc 0.990 spect	0.0841 (3%, eK/γ 1.6, K/L/M $\approx$ 1/2.6/0.75) spect conv, β - γ coinc γ (~3%, e/γ 9.4, K/L + M 0.22) α - γ coinc, scint spect 0.085 (K/L 0.16, L _I /L _{II} /L _{III} $\approx$ <0.1/0.83/1.00) spect conv

Isotope <i>Z A</i>	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
				Particles	Gamma transitions
¹⁷⁰ Tm (Cont.)					0.085 (e_K/γ 1.5) scint spect 0.084 (e/γ 4, $K/L/M$ $\sim 1/6.9/2.1$) spect, spect conv, β - γ coinc γ ($e/\gamma > 10$, $K/L +$ M 0.1) spect conv, scint spect 0.085 spect conv
^{171m} Tm		<i>IT</i>	2.5×10^{-6} s delay coinc		0.113 (e/γ 1.3) spect conv, β - γ coinc 0.113 (e/γ large) spect conv, β - γ coinc
¹⁷¹ Tm		β^-	680 d	0.10 spect	
^{172(?)} Tm		β^-	2-3 d		
¹⁷¹ Tm			19 m		
¹⁶⁸ Yb		<i>EC</i>	62 h		
¹⁶⁸ Yb	0.140				
¹⁶⁹ Yb		<i>EC</i>	31.8 d 32.4 d 33 d		γ_1 0.023, γ_2 0.064, γ_3 0.095, γ_4 0.110 (e/γ 1.6), γ_5 0.120, γ_6 0.133 (e/γ 0.2), γ_7 0.143, γ_8 0.160, γ_9 0.178 (e/γ 0.8), γ_{10} 0.198 (e/γ 0.4), γ_{11} 0.308 (e/γ 0.04); ($\gamma_2/\gamma_4/\gamma_5/$ $\gamma_9/\gamma_{10}/\gamma_{11} \approx$ 1.3/2.1/2.0/1.0/ 1.7/0.6) spect, spect conv, delay coinc 0.063, 0.094, 0.110, 0.131, 0.177, 0.198, 0.308 spect conv 0.109, 0.130, 0.177, 0.198, 0.307 scint spect, delay coinc
^{170m} Yb		<i>IT</i>	1.57×10^{-9} s delay coinc 1.6×10^{-9} s delay coinc		0.0841 (e_K/γ 1.6, $K/L/M \approx 1/2.6/$ 0.75) spect conv, β - γ coinc 0.085 (e_K/γ 1.5) scint spect
¹⁷⁰ Yb	3.03				
¹⁷¹ Yb	14.31				
¹⁷² Yb	21.82				
¹⁷³ Yb	16.13				

Isotope Z	A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
					Particles	Gamma transitions
⁷⁰ Yb ¹⁷⁴		31.84				
Yb ¹⁷⁶			β^-	102 h 99 h 101 h	0.50, 0.13 abs 0.45 cl ch	0.138, 0.259, 0.283, 0.396 spect conv Others
Yb ¹⁷⁶		12.73				
Yb ¹⁷⁷			β^-	1.8 h 1.9 h 2.4 h	1.3 abs 1.2 cl ch	0.150 (K/L 3) spect conv, β - γ coinc
Yb ^m			IT	6 s		0.212, 0.104 (?) scint spect 0.200 abs Yb K- α
Yb ^m			IT	50 s		~ 0.025 , Yb L- α abs
Yb ^m				0.15 s		0.455 scint spect
⁷¹ Lu ¹⁷⁰			EC	1.7 d		~ 2.5 abs
Lu ¹⁷¹			EC	8.5 d		~ 1.2 abs
Lu ¹⁷¹			EC	~ 600 d		~ 1 abs
Lu ¹⁷²			EC	6.70 d		1.2 abs
Lu ¹⁷²			β^+ , EC (?)	4.0 h	1.2 abs	
Lu ¹⁷³			EC	~ 500 d		~ 0.2 , 0.8 abs
Lu ¹⁷⁴			EC $\sim 80\%$, $\beta^- \sim 20\%$	165 d	β^- : 0.6 abs	~ 1 abs
Lu ¹⁷⁵		97.40				
Lu ^{176m}			β^- , no IT	3.67 h 3.7 h	1.1, 1.2 1.3 cl ch 1.2 abs	0.0889 (K/L ₁₁ /L ₁₁₁ $\approx 0.24/0.71/$ 1.00) spect conv 0.089 (e_K/γ 1.3) scint spect, β - γ delay coinc 0.089 (e/γ large, K/L 0.1) scint spect
Lu ¹⁷⁶		2.60	β^- , no EC	7.5×10^{10} y sp act	0.40 abs	0.089, 0.180, 0.270 scint spect
Lu ^{177m}			IT	1.3×10^{-7} s delay coinc		0.150 (K/L 3) scint spect, β - γ coinc
Lu ¹⁷⁷			β^-	6.8 d 7.0 d 6.6 d	0.495 (65%), 0.37 (17%), 0.17 (18%) spect 0.475 spect	0.112 (e_K/γ 0.81), 0.206 (e_K/γ 0.04), 0.318 ($\sim 5\%$) scint spect, γ - γ coinc 0.112, 0.206, 0.317 (very weak) spect, spect conv 0.112 (K/L/M $\approx$ 1/2/0.5), 0.205 spect conv 0.113, 0.209 spect conv Others

Isotope Z A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
				Particles	Gamma transitions
^{171}Lu ^{171, 179}			22 m		
^{172}Hf ¹⁷⁰		β^+	112 m	2.4 spect	No γ
Hf^{171}		EC	16.0 h		1.4 abs
Hf^{172}		EC	~ 5 y		$\sim 0.28, 0.8$ abs, spect conv
Hf^{173}		EC	23.6 h		~ 1 abs
Hf^{174}	0.18				
Hf^{175}		EC	70 d		0.089, 0.113, 0.228, 0.318, 0.342 (K/L 4.9), 0.431 spect conv 1.5 abs
Hf^{176m}		IT	1.35×10^{-9} s delay coine		0.089 (e/γ 1.3) scint spect, β - γ coine
Hf^{176}	5.15				
Hf^{177}	18.39				
Hf^{178}	27.08				
Hf^{179m}		IT	19 s		0.160, 0.217 scint spect, spect conv $\sim 0.150, 0.215$ scint spect, conv- γ coine 0.150 (e/γ very large, K/L 0.9) spect conv 0.220 scint spect
Hf^{179}	13.78				
Hf^{180m}		IT	5.5 h		0.057, 0.093, 0.214, 0.330, 0.442 spect conv, γ -conv coine
Hf^{180}	35.44				
Hf^{181}		β^-	45 d 47 d	0.408 spect 0.420 spect 0.410 spect 0.404 spect 0.460 spect	γ_1 0.133, γ_2 0.136, γ_3 0.344, γ_4 0.481, γ_5 0.611 spect conv, β -conv, conv-conv, β - γ coine γ_1 0.133, γ_2 0.136, γ_3 0.345, γ_4 0.481, γ_5 0.615 spect conv γ_1 0.133 (K/L ~ 1), γ_2 0.136 (K/L ~ 0.2), γ_3 0.345, γ_4 0.481, γ_5 0.612 (γ_1 and γ_2 coine with γ_3 , γ_1 coine with γ_4 , γ_4 not coine with γ_3) spect conv, conv- conv coine

Isotope Z A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
				Particles	Gamma transitions
^{181}Hf (Cont.)					γ_1 (eK/γ 0.51), γ_4 (eK/γ 0.034) scint spect, γ - γ coinc γ_1 0.130 (e/γ 0.90, $K/L + M$ 0.6), γ_2 0.134 ($e/\gamma \sim 3$, $K/L + M \sim 8$), γ_3 0.340 ($e/\gamma \sim 0.1$), γ_4 0.474 (e/γ 0.030, $K/L + M$ 4.0) spect conv γ_1 0.132, γ_2 0.135 ($K/L \sim 8$), γ_3 0.345, γ_4 0.481 ($\gamma_1/\gamma_2 \sim 5$, $\gamma_4/\gamma_3 \sim 8$) spect, spect conv γ_1 0.130 ($eK/\gamma \sim 1.3$), γ_2 0.134, γ_3 0.337 ($K/L \sim 3.6$), γ_4 0.471 ($K/L \sim 3$) ($\gamma_3/\gamma_4 \sim 2.5$) spect conv, β - γ , conv-conv coinc γ_3 0.347 ($K/L \sim 5.0$), γ_4 0.485 ($K/L \sim 5.2$) spect, spect conv
Hf^m		IT	~ 3.5 s		
^{176}Ta		EC	8.0 h	Conv: 0.1, 0.2, ~ 1 abs	~ 2 abs
Ta^{177}		EC	53 h	Conv: 0.1 abs	~ 1.4 (weak) abs
Ta^{178}		EC $\sim 97\%$, $\beta^+ \sim 3\%$	2.1 h	β^+ : ~ 1 abs Conv: ~ 0.1 abs	1.3-1.5 abs
Ta^{178}		EC $\sim 94\%$, $\beta^+ \sim 6\%$	9.35 m	β^+ : 1.06 spect Conv: 0.08 spect conv	~ 1.5 abs
Ta^{179}		EC	~ 600 d	Conv: ~ 0.1 abs	~ 0.7 (weak) abs
Ta^{180}		EC $\sim 79\%$, $\beta^- \sim 21\%$, no β^+ (lim 0.005%)	8.15 h 8.00 h 8.2 h	0.71 ($\sim 50\%$), ~ 0.61 ($\sim 50\%$) spect 0.7 spect, abs	γ_1 0.093 ($K/L \sim 0.15$), γ_2 0.102 ($\gamma_1 + \gamma_2: e/\gamma \sim 5$), γ_3 0.2, γ_4 0.4 (γ_3 and γ_4 very weak) spect conv, scint spect, β - γ , x - γ coinc 1.3 abs
Ta^{181m_2}		IT	2.2×10^{-8} s delay coinc 2.0×10^{-8} s delay coinc		See γ 's of Hf^{181}
Ta^{181m_1}		IT	1.2×10^{-8} s delay coinc 1.1×10^{-8} s delay coinc 1×10^{-8} s delay coinc		See γ 's of Hf^{181}
Ta^{181}	100				

Isotope Z A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
				Particles	Gamma transitions
$n\text{Ta}^m$		IT	0.33 s		Ta L-x
Ta^{182m}		IT	16.5 m	β^- : 0.6 abs	0.180 (K/L 0.25) spect conv
		IT ~95%, β^- ~5%	16.2 m		0.180 (eK/ γ 0.8) scint spect
Ta^{182}		β^-	111 d	0.525, other β^- 's, spect	γ_1 0.065714, γ_2
			113 d	0.53 spect	0.067736, γ_3
			117 d		0.084667, γ_4 0.10009, γ_5 0.11366, γ_6 0.11640, γ_7 0.15241, γ_8 0.15637, γ_9 0.17936, γ_{10} 0.19831, γ_{11} 0.22205, γ_{12} 0.22927, γ_{13} 0.26409, γ_{14} 1.121, γ_{15} 1.88, γ_{16} 1.223 (rel intens: γ_1 9, γ_2 100, γ_3 6, γ_4 46, γ_5 9, γ_6 2, γ_7 43, γ_8 14, γ_9 19, γ_{10} 9, γ_{11} 45, γ_{12} 24, γ_{13} 27, γ_{14} 352, γ_{15} 157, γ_{16} 334) cryst spect 0.046, 0.058, 0.065, 0.067, 0.075, 0.077, 0.084, 0.100, 0.113, 0.134, 0.143, 0.152, 0.178, 0.198, 0.221, 0.228, 0.245, 0.262 spect conv 1.121, 1.189, 1.219 spect, spect conv 0.082, 0.098, 0.112, 0.122, 0.132, 0.141, 0.157, 0.165, 0.172, 0.198, 0.222, 0.243, 0.255, 0.264, 0.290 (?), 0.299 (?), 0.324, 1.133, 1.219, 1.237 spect, spect conv 0.224, 0.232, 0.260, 0.268, 0.280, 0.320, 0.342, 0.362, 0.392, 0.412, 0.421, 0.526, 0.565, 0.607, 0.624, 0.728, 0.763, 0.780, 0.892, 0.935, 0.993, 1.133, 1.215, 1.231 spect

Isotope <i>Z</i> <i>A</i>	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
				Particles	Gamma transitions
⁷³ Ta ¹⁸³		β^-	5.2 d 6.0 d 6.1 d	0.65 scint spect 0.6 abs	0.24 scint spect γ
Ta ¹⁸⁴		β^-	9.3 h	1.4 abs	0.410, 0.86, 1.10 scint spect
Ta ¹⁸⁵		β^-	48 m	1.6, 0.15 (conv ?) abs 1.7 abs	
⁷⁴ W ¹⁷⁶		EC 99+%, β^+ ~0.5%	80 m	β^+ : ~2 abs Conv: ~0.1, ~0.2 abs	~1.3 abs
W ¹⁷⁷		EC	130 m	Conv: 0.13, ~0.4 abs	~0.5, 1.2 abs
W ¹⁷⁸		EC	21.5 d		~0.3 (weak) abs
W ¹⁷⁹		EC	30 m		
W ¹⁷⁹		EC or IT	5.2 m		
W ¹⁸⁰	0.135				
W ¹⁸¹		EC	140 d		0.030, 0.600, 0.800 scint spect 1.8 (weak) abs
W ¹⁸²	26.4				
W ^{183m}		IT	5.5 s	Conv: 0.08 abs	0.12, 0.17 scint spect
W ¹⁸³	14.4				
W ¹⁸⁴	30.6				
W ^{185m}		IT	1.85 m	Conv: 0.075 scint spect	
W ¹⁸⁶		β^-	73.2 d 75 d	0.428 spect 0.43 spect Others	No γ Others
W ¹⁸⁶	28.4				
W ¹⁸⁷		β^-	24.1 h 24.0 h	1.33 (30%), 0.63 (70%) spect 1.32 ($\geq 12\%$), 0.63 ($\geq 65\%$), ~0.38 ($\leq 23\%$) spect 1.34, ~0.65 spect Others	0.07200, 0.13425, 0.4795, 0.6189, 0.6861 cryst spect γ_1 0.072 (coinc with γ_2 , delay coinc with γ_2), γ_2 0.134 (coinc with γ_1 , γ_4 and γ_7 , delay coinc with γ_2), γ_3 0.480 ($e\pi/\gamma$ 0.022, delay coinc with γ_1 and γ_2 , not coinc with γ_4 or γ_6), γ_4 0.552 (coinc with γ_2), γ_5 0.618 (not coinc with other γ 's), γ_6 0.69, γ_7 0.775 (coinc with γ_2) ($\gamma_2/\gamma_3/\gamma_4/\gamma_5/\gamma_6$ /' γ_7 = 0.45/1.00/ 0.31/0.42/1.48/ 0.23) scint spect

Isotope Z	A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
					Particles	Gamma transitions
^{187}W (Cont.)						0.133, 0.204, 0.478, 0.615, 0.680, 0.767 spect conv 0.129, 0.462, 0.652 spect conv, β - γ , γ - γ coinc 0.078, 0.138 spect conv Others
^{188}W			β^-	65 d genet		
^{182}Re			EC	12.7 h 14 h		0.110, 0.127, 0.222, 0.250, 0.346 spect conv, spect
^{182}Re			EC	64.0 h 67 h		0.110, 0.127, 0.222, 0.250, 0.346 spect conv, spect
^{183}Re			EC	155 d 120 d		0.081, 0.252 spect conv
^{184}Re			EC	50 d		0.159, 0.206, 0.244, 0.784, 0.80 spect conv 0.043, 0.159, 0.205, 0.285 spect, spect conv 1.0 abs
^{184}Re			EC or IT	2.2 d		0.043, 0.159 spect, spect conv
^{185}Re	37.07					
^{186}Re			β^- ~95%, EC ~5%, no β^+ (lim 10 ⁻⁶ %) β^- ~91%, EC ~9%	92.8 h 91 h 90 h	1.07 (80%), 0.93 (20%) spect 1.070 (73%), 0.942 (27%) spect 1.090 (67%), 0.95 (30%), 0.64 (3%) spect 1.063 spect 1.07 spect	With β^- : γ_1 0.137 (e_K/γ ~0.35, $K/L/M$ = 0.6/1/0.2), 0.627, 0.764 spect, spect conv, β - γ , γ - γ coinc γ_1 0.136 (e_K/γ 0.37, $K/L/M$ = 0.6/1/0.2) spect, spect conv, β - conv, γ - γ coinc With EC: γ_2 0.123 (~2%), γ_1/γ_2 = 9 γ_2 0.122 (3%, e_K / γ 0.45, K/L 0.6)
^{187m}Re			IT	5.3×10^{-7} s delay coinc 5.5×10^{-7} s delay coinc		0.133 (e_K/γ $\leq$ 3.2, K/L ~5) scint spect See γ 's of ^{187}W

Isotope Z A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
				Particles	Gamma transitions
^{187}Re	62.93	β^-	4×10^{13} y sp act	0.400 ion ch 0.043 abs	No γ , no x
$\text{Re}^{188\text{m}}$		<i>IT</i>	22 m 17 m		0.0635, 0.092, 0.106 spect conv, scint spect
Re^{188}		β^-	16.9 h 18.9 h 18 h	2.07 (coinc with 0.152 γ) spect, β - γ coinc 2.10 spect 2.05 abs Others	0.152 (70%, eK/γ 0.05, K/L 0.42), 0.476 (3%), 0.638 (6%), 0.933 (5%), 1.3 (5%) spect, abs, spect conv 0.15, 0.48, 0.64, 0.95, 1.40 spect 0.16, 0.48, 0.64, 0.94, 1.43 spect 1.39 coinc abs 0.154 spect conv
Re^{189}		β^-	150 d 250-300 d	0.2 abs	1.0 abs
Re		β^-	≥ 5 y	0.75 abs	
^{182}Os		<i>EC</i> , no β^+	24.0 h		
Os^{183}		<i>EC</i>	12.0 h	Conv: 0.15, 0.42 spect conv	0.3, 1.6 abs
Os^{184}	0.018				
Os^{186}		<i>EC</i> (L/K ~ 0.35) No β^+	97 d 95 d		γ_1 0.648, γ_2 0.878 ($\gamma_1/\gamma_2 \sim 6$) spect, γ - γ coinc γ_1 0.65, γ_2 0.88 (γ_1/γ_2 6.1) scint spect 0.235, 0.653 spect conv
$\text{Os}^{186\text{m}}$		<i>IT</i>	8×10^{-10} s delay coinc		0.137 scint spect
Os^{186}	1.59				
$\text{Os}^{187\text{m}}(?)$			35 h		
Os^{187}	1.64				
Os^{188}	13.3				
Os^{189}	16.1				
$\text{Os}^{190\text{m}}(?)$			6 h		
$\text{Os}^{190\text{m}}$			9.5 m		
Os^{190}	26.4				

Isotope Z A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
				Particles	Gamma transitions
^{191}Os		IT , no β^- lim 5%	14 h		0.0742 ($L_1/L_{II}/L_{III}$ $\approx 42/24/100$) spect conv
^{191}Os		β^-	16.0 d 16.1 d 15.0 d	0.143 spect 0.142 spect	γ_1 0.0417 (L_{II}/L_{III} $\approx 32/40$, e/γ large), γ_2 0.129 (coinc with γ_1 , $K/L_1/L_{II}/L_{III} \approx$ 100/30/11/6) spect conv, conv- conv coinc γ_1 (eK/γ 1.36) 0.042 (L/M 1.8), 0.128 ($eK/\gamma \sim 0.5$, K/L 2.1) spect conv 0.041, 0.128 spect conv, spect 0.039, 0.127 spect conv 0.129 spect conv
^{192}Os	41.0				
^{193}Os		β^-	30.6 h 31.9 h 32 h 30 h	1.10 spect 1.05 scint spect 1.15 abs	With Ir^{193m} : 0.066 spect conv 0.065 scint spect, β - γ delay coinc No γ
^{194}Os		$[\beta^-]$	~ 700 d		
^{187}Ir		EC 99+%, $\beta^+ \sim 0.2\%$	11.8 h	β^+ : 2.2 spect Conv: 0.3, 1.2 spect conv	~ 1.3 abs
^{188}Ir		EC 99+%, $\beta^+ \sim 0.3\%$	41.5 h	β^+ : 2.0 spect Conv: 0.2, 0.9 spect conv	~ 1.8 abs
^{190}Ir		β^+ , EC (?)	3.2 h	β^+ : 1.7 spect Conv: 0.2, 0.8 spect conv	
^{190}Ir		EC	12.6 d 10.7 d		0.2, 0.6 abs 0.3 abs
^{191}Ir	38.5				
^{192m}Ir		IT	1.42 m 1.5 m		0.057 spect conv 0.056 spect conv γ (continuum) scint spect γ ($eL/\gamma > 400$) ion ch
^{192}Ir		EC , β^- No β^+ (lim 0.008%)	74.37 d 74.5 d 74.7 d	0.66 spect 0.67 spect 0.68 coinc abs 0.6 abs	γ_1 0.13633, γ_2 0.20131, γ_3 0.20574, γ_4 0.29594, γ_5 0.30845, γ_6 0.31646, γ_7 0.46798, γ_8 0.4848, γ_9 0.5884,

Isotope <i>Z A</i>	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
				Particles	Gamma transitions
$^{192}_{77}\text{Ir}$ (<i>Cont.</i>)					γ_{10} 0.6045, γ_{11} 0.6129 (rel abund γ_1 4, γ_2 10, γ_3 75, γ_4 380, γ_5 370, γ_6 990, γ_7 300, γ_8 11, γ_9 11, γ_{10} 14, γ_{11} 5) cryst spect 0.136, 0.151 (or 0.156), 0.169 (or 0.173), 0.201, 0.206, 0.283, 0.295, 0.308, 0.316, 0.396 (or 0.400), 0.415, 0.434 (or 0.438), 0.467, 0.484, 0.589, 0.604, 0.611 spect conv 0.775, 0.870 scint spect Others
$\text{Ir}^{193\text{m}}$		<i>IT</i>	5.7×10^{-9} s delay coine		0.065 scint spect, β - γ delay coine
Ir^{193}	61.5				
Ir^{194}		β^-	19.0 h 19.5 h 19 h	2.18 spect 2.2 spect 2.1 abs 0.5 β - γ coine abs	γ_1 0.290, γ_2 0.326, γ_3 1.51 ($\gamma_1/\gamma_2/\gamma_3 \approx$ 1/5/1) spect 0.328 spect conv 1.7-2.2 (0.14%) Be. <i>D</i> - γ - n reaction Others
Ir^{196}		β^-	140 m	1.8 abs ~1 abs	0.49, 0.84 scint spect
Ir^{196}		β^-	9 d	~0.08 abs	γ
Ir^{197}		β^-	7 m	1.65, 0.6 abs	γ
Ir^{198}		β^-	45 s	3.6 abs	0.78 scint spect
$^{190}_{78}\text{Pt}$	0.012				
Pt^{191}		<i>EC</i>	3.00 d		0.083, 0.096, 0.173 spect 0.6, 1.5 abs, abs conv
Pt^{192}	0.78				
$\text{Pt}^{193\text{m}}$		<i>IT</i> <i>EC</i>	4.33 d 4.6 d		0.135 (<i>K/L</i> 0.28, <i>L</i> _I / <i>L</i> _{II} / <i>L</i> _{III} 1/0/2) spect conv 0.2, 1.5 abs
Pt^{194}	32.8				

Isotope Z A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
				Particles	Gamma transitions
^{196m}Pt		<i>IT</i>	3.5 d 3.8 d 4.4 d 3.3 d		0.029 ($e/\gamma > 7.5$), 0.097 (e/γ 9.0, K/L 5.7), 0.126, 0.129 (e/γ very large, K/L 0.26) spect, spect conv, γ - conv coine 0.099, 0.130 (K/L 0.1) spect conv 0.126 (K/L 0.23) spect conv
Pt^{196}	33.7				
Pt^{196}	25.4				
Pt^{197m}		<i>IT</i>	78 m 80 m 88 m		0.337 (e/γ very large, K/L 1.3) spect conv
Pt^{197}		β^-	18 h 17.4 h	0.670 spect 0.7 abs	0.077, 0.191 (K/L 6.0) spect conv 0.077, 0.191 spect, spect conv
Pt^{198}	7.23				
Pt^{199}		β^-	31 m	1.8 abs	
Pt		β^-	82 d	0.5 abs	0.6 abs
$^{183-187}\text{Au}$		<i>EC</i> , β^+ , α ~0.01%	4.3 m	α : 5.07 ion ch	
Au^{191}		<i>EC</i>	18 h ~1 d		0.053, 0.064, 0.111, 0.123, 0.166, 0.250, 0.405 spect conv
Au^{192}		<i>EC</i> , β^+	5.0 h 4.7 h 4.1 h	β^+ : ~1.9 abs Conv: ~0.4 abs	2-3 abs
Au^{193}		<i>EC</i>	15.8 h 15.3 h		0.051, 0.060, 0.084, 0.093, 0.109, 0.165, 0.177, 0.235 spect conv
Au^{194}		<i>EC</i> ~97%, β^+ ~3%	39.5 h 39 h	1.8 abs	0.291 (e/γ 0.054, K/L 2), 0.328 (e/γ 0.19, K/L 2), 0.466 (weak) 1.48 (e/γ 0.0026), 2.1 spect, spect conv
Au^{196m}		<i>IT</i>	0.5 m		0.056, 0.259 spect conv
Au^{196}		<i>EC</i>	180 d 185 d		0.0308, 0.0990, 0.130 spect conv 0.029 (L/M 4.6), 0.097 (K/L 5.8), 0.126 spect, spect conv, γ -conv coine Others
Au^{198}		<i>EC</i> or <i>IT</i>	14.0 h 13 h		

Isotope Z A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
				Particles	Gamma transitions
⁷⁹ Au ¹⁹⁶		EC ~95%, β ⁻ ~5%	5.55 d	β ⁻ :	With EC:
		EC ~80%, β ⁻ ~20%	5.60 d	0.27 spect 0.30 spect 0.84 abs	0.332, 0.354 spect conv 0.330 (K/L 1.7), 0.358 (K/L 1.7) spect conv With β ⁻ : 0.426 (e _L /γ 0.007, L/M + N 3.5), no 0.175 γ spect conv, β-conv coinc 0.175 spect conv
^{197m} Au		IT	7.4 s		0.130 (e _K /γ ≤2.0, K/L/M ≈ 1/7.5/ 3.6), 0.279 (e _K /γ ~0.3) spect conv
			7.5 s		0.273 spect conv, conv-conv coinc 0.275 scint spect 0.25 abs conv
¹⁹⁷ Au	100				
¹⁹⁸ Au		β ⁻	2.69 d	β ₁ 0.963 spect	γ ₁ 0.41177 cryst spect
		No EC (lim 0.2%)	2.73 d	β ₂ 0.290 (coinc with 0.680 γ) β-γ coinc spect	γ ₁ 0.4116 spect
		No EC (lim 0.4%)	2.66 d	β ₃ 1.37 (0.01%) spect	γ ₁ (e _K /γ 0.031, K/L 3.1, L/M 3.3) spect conv
		No β ⁺ (lim 0.003%)			γ ₁ (e _K /γ 0.03, K/L 3, L/M 3.3) spect, spect conv
					γ ₁ (e _K /γ 0.029, K/L 2.1, L/M 4.3) spect conv
					γ ₁ (e _K /γ 0.026 K/L 2.2, L/M 4) spect conv
					γ ₁ (e _K /γ 0.03, K/L + M 2.3) spect conv
					γ ₁ (L _{III} /L _{III} 2.5) spect conv
					γ ₂ 0.676 (0.5%, e _K /γ 0.034, K/L 5.7), γ ₃ 1.088 (0.1%, e _K /γ 0.005, K/L 6.3) spect, spect conv
					γ ₂ 0.680 (1%, coinc with 0.411 γ and 0.29 β ⁻), γ ₃ 1.09 (~0.2%) scint spect, β-γ coinc
					γ ₂ 0.673 (1.4%), γ ₃ 1.08 (0.3%) spect γ ₂ 0.67 (1.4%, coinc with 0.411 γ), γ ₃ 1.09 (0.3%) scint spect, γ-γ coinc

Isotope Z A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
				Particles	Gamma transitions
^{199}Au		β^-	3.15 d 3.2 d 3.3 d	0.460, 0.297 (coinc with 0.158 γ), 0.250 (coinc with 0.207 γ) β - γ spect coinc ~ 0.47 β^- (not coinc with γ) β - γ coinc 0.47 ($\sim 4\%$), 0.30, 0.25 spect 0.443 ($\sim 7\%$), 0.291 spect 0.32 spect abs Others	γ_1 0.050 (eL/γ 6), γ_2 0.159 (eK/γ 0.19, K/L 0.6), γ_3 0.209 (eK/γ 0.54, K/L 5.4) (γ_1/γ_2 / $\gamma_3 \approx 0.84/100$ / 23.8) spect, spect conv, γ - γ coinc γ_1 (coinc with γ_2), γ_3 (not coinc with γ_1 or γ_2) scint spect, γ - γ coinc 0.050, 0.158 (eK/γ 0.24, K/L 0.73, L/M 3.3), 0.208 (eK/γ 0.62, K/L ~ 5.6 , $L/M \sim 4$) spect, spect conv, scint spect, γ - γ , β - γ coinc 0.0198, 0.159 (K/L 0.56, L/M 3.6), 0.208 (K/L 4.5) spect conv Others
^{200}Au		β^-	48 m	2.2 abs ~ 2.5 abs	1.13, 0.39 scint spect
^{201}Au		β^-	26 m	1.5 abs	0.55 scint spect
$^{202,204}\text{Au}$		β^- or IT	25 s		
^{203}Au		β^-	55 s	1.9 abs	0.69 scint spect
^{196}Hg		α	0.7 m	α : 5.60 ion ch	
^{189}Hg			30 m		
^{190}Hg			90 m		
^{191}Hg		EC	12.4 h		
^{192}Hg		EC β^+	5.7 h 8.4 h	1.18 spect	1.4 abs
^{193}Hg		EC	10.0 h 14.5 h (?), 29.0 h (?)		
^{196m}Hg		EC, IT	38 h ~ 31 h 40 h		0.036, 0.056 (coinc with Au^{196m}), 0.122, 0.259 (coinc with Au^{196m}) spect conv 0.037, 0.056, 0.122 (conv in Hg), 0.206, 0.261, 0.318, 0.558 spect conv
^{198}Hg		EC	9.5 h		0.061, 0.179, 0.600, 0.780 spect conv 0.061, 0.179 spect conv

Isotope Z A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
				Particles	Gamma transitions
^{196}Hg	0.146				
$\text{Hg}^{197\text{m}_2}$		<i>IT</i> 97%, <i>EC</i> 3%	23 h 25 h		With <i>IT</i> : 0.133 (e_K/γ 0.5, $K/L/M + N \approx$ 1.0/2.4/0.8), 0.164 (e_K/γ 4.6, $K/L/M + N \approx$ 1.0/2.3/1.2) spect conv, γ - γ coinc 0.134 ($K/L < 0.1$), 0.165 (K/L ~ 0.05) spect conv 0.134 ($L_I/L_{II}/$ $L_{III} \approx 0.05/$ 1.1/1.0), 0.165 ($L_I/L_{II}/L_{III} \approx$ 1.0/ $< 0.1/1.5$) spect conv With <i>EC</i> : 0.191 ($e_K/\gamma \sim 1.7$, $K/L \sim 6$), 0.275 (weak, e_K/γ ~ 0.5 , $K/L \sim 5$) spect conv, γ - γ coinc
$\text{Hg}^{197\text{m}_1}$		<i>IT</i>	7.0×10^{-9} s delay coinc 8×10^{-9} s delay coinc		0.13 scint spect
Hg^{197}		<i>EC</i>	65 h 66 h 64 h		0.077 (e_L/γ 2.5, L/M 3.6), 0.191 (e_K/γ ~ 1.7 , $K/L \sim 6$) spect conv, γ - γ coinc 0.077 ($L_I/L_{II}/L_{III}$ $\approx 1.0/0.45/0.34$) spect conv 0.078 (L_I/M 4), 0.191 ($K/L \sim 9$) spect conv 0.077, 0.278 spect conv
Hg^{198}	10.02				
$\text{Hg}^{199\text{m}_2}$		<i>IT</i>	44 m 43 m		0.155 (e/γ 0.25, K/L < 0.4), 0.368 (e/γ > 11 , K/L 1.6) spect conv 0.159 (L_{II}/L_{III} 1.6) spect conv 0.16 γ (coinc with 0.37 γ) scint spect, γ - γ coinc Others

Isotope <i>Z A</i>	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
				Particles	Gamma transitions
$^{199m}_{80}\text{Hg}$		<i>IT</i>	2.4×10^{-9} s delay coinc		0.158 spect conv, β - γ coinc
Hg^{199}	16.84				
Hg^{200}	23.13				
Hg^{201}	13.22				
Hg^{202}	29.80				
Hg^{203}		β^-	47.9 d 45.9 d 46.5 d 43.5 d	0.208 β - γ coinc spect 0.210 spect 0.205 spect Others	0.279 (e/γ 0.27, <i>K/L</i> 3) spect, spect conv, β - γ coinc 0.278 (e_K/γ 0.19, <i>K/L + M</i> 3.7) spect, spect conv ~ 0.28 (e_K/γ 0.23) scint spect 0.279 (<i>K/L</i> ~ 10) spect conv 0.286 (e/γ 0.3, <i>K/L</i> 3) spect conv Others
Hg^{204}	6.85				
Hg^{205}		β^-	5.5 m 5.6 m	1.8 abs 1.6 abs	
$^{198}_{81}\text{Tl}$		<i>EC</i>	1.8 h	Conv: ~ 0.4 abs	Several γ 's, abs
Tl^{199}		<i>EC</i> No β^+	7 h		0.049, 0.078, 0.103, 0.157, 0.206, 0.245, 0.332, 0.454, 0.490 spect conv, γ - γ coinc
Tl^{200}		<i>EC</i> No β^+	27 h		0.365, 0.577, 0.622, 0.829, 1.210, 1.360 spect conv ~ 0.4 , 1.6 abs
Tl^{201}		<i>EC</i>	72 h		0.210 spect conv, abs
Tl^{202}		<i>EC</i> <i>EC (L/K</i> $\sim 1)$, no β^+ or β^-	12.5 d 11.5 d 11.8 d		0.435 spect conv, abs 0.431 scint spect
Tl^{203}	29.50				
Tl^{204}		β^- $\sim 98\%$, <i>EC</i> $\sim 2\%$, β^- $\sim 98.5\%$, <i>EC</i> $\sim 1.5\%$	3.5 y 2.7 y	0.765 spect 0.760 scint spect 0.783 spect 0.77 spect	No γ (lim 0.01%) No γ (lim 0.5%) No γ Hg <i>K-x</i>
Tl^{205}	70.50				
Tl^{206}		β^-	4.19 m 4.23 m 4.3 m	1.51 spect 1.65 abs 1.8 abs No conv	No γ

Isotope Z A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
				Particles	Gamma transitions
$_{81}\text{Tl}^{207}$ (AcC')		β^-	4.79 m 4.77 m 4.76 m	1.44 abs 1.47 abs 1.6 abs	0.870 ($\sim 0.5\%$)
Tl^{208} (ThC')		β^-	3.1 m	1.792 spect 1.795 spect 1.805 spect, $\beta\text{-}\gamma$ coinc 1.72 spect 1.82 abs $\beta^- > 1.792$ (?) ($< 1\%$) spect	2.62 ($\sim 100\%$, e/γ ~ 0.002), 0.859 ($\sim 15\%$, e/γ ~ 0.02), 0.582 ($\sim 80\%$, e/γ ~ 0.02), 0.510 ($\sim 25\%$, e/γ ~ 0.08), 0.277 ($\sim 10\%$, e/γ ~ 0.3) spect, spect conv 2.6143 spect, 0.51085 spect 2.6147 spect conv 2.616, 0.510 spect 2.615 spect 2.58 (100%), 0.575 (100%), 0.51 (50%) spect 2.62, 0.582, 0.510, 0.277 spect conv 2.62, 0.868, 0.581, 0.502 spect 2.62 scint spect No 3.2 γ
Tl^{209}		β^-	2.2 m	1.99 spect 1.8 abs	0.12 scint spect
Tl^{210} (RaC')		β^-	1.32 m 1.5 m ~ 1.3 m	1.8 cl ch 2.0 abs	No $\gamma > 2.8$, $D\text{-}\gamma\text{-}p$ reaction
$_{82}\text{Pb}^{198}$		EC	25 m		
Pb^{199}		EC	~ 80 m		
Pb^{200}		EC	18 h		0.139, 0.320 spect conv
Pb^{201m}		IT	50 s		0.25, 0.42, 0.67 scint spect
Pb^{201}		EC	8 h		
Pb^{202m}		IT	5.6 s		0.89 scint spect
Pb^{202}	$< 4 \times 10^{-4}$		> 500 y genet, yield		
Pb^{203}		EC	52 h 54 h		0.153, 0.269, 0.422 spect conv 0.270, 0.420 abs conv 0.270, ~ 0.470 spect, spect conv

Isotope Z A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
				Particles	Gamma transitions
$^{204m}_82\text{Pb}$		<i>IT</i>	68 m 65 m		0.905 ($e/\gamma \sim 0.1$, K/L 1.5), 0.374 (with $^{204m}_81\text{Pb}$, e/γ ~ 0.05 , K/L 2.1) spect conv, abs conv, abs 0.90 abs conv 1.1 abs conv, abs
$^{204m}_81\text{Pb}$		<i>IT</i>	3×10^{-7} s delay coinc		0.374 (see γ 's of $^{204m}_82\text{Pb}$)
$^{204}_82\text{Pb}$	1.48				
$^{206}_82\text{Pb}$	23.6				
$^{207m}_82\text{Pb}$		<i>IT</i>	0.84 s 0.82 s 0.80 s		0.55, 1.05 scint spect 0.5, 1.1 scint spect
$^{207}_82\text{Pb}$	22.6				
$^{208}_82\text{Pb}$	52.3				
$^{209}_82\text{Pb}$		β^-	3.22 h 2.75 h	0.635 spect 0.620 spect 0.68 spect	No γ , no conv No γ
$^{210}_{82}\text{Pb}$ (RaD)		β^-	22 y	0.018 ion ch 0.018 scint spect, β - γ coinc 0.017 ($\leq 90\%$), 0.056 ($\geq 10\%$) ion ch Others	γ_1 0.0465, no other γ between 0.016 and 0.060 (lim $\sim 2\%$ of γ_1) cryst spect γ_1 0.0467, no other γ (lim 5% of γ_1) cryst spect 0.0467 (3.5%) spect conv, abs γ_1 ($eL/\gamma \sim 16$, $L_1/L_{11}/L_{111}/M \approx$ 1.0/0.09/0.019/ 0.29) spect conv γ_1 ($e/\gamma \sim 23$) spect conv γ_1 ($e/\gamma \sim 17$, using $\gamma = 3.5\%$) spect conv 0.032, 0.037, 0.0467 cryst spect 0.065 ($< 0.2\%$), 0.0467 (2.8%), 0.043 (0.2%), 0.037 (0.2%), 0.032 (0.4%), 0.023 ($\sim 1\%$), 0.007 ($\sim 10\%$) cryst spect, cl ch, abs Others

Isotope <i>Z</i> <i>A</i>	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
				Particles	Gamma transitions
$^{211}_{82}\text{Pb}$ (AcB)		β^-	36.1 m 36.0 m	1.39 (~80%), ~0.5 (~20%) abs	0.065, 0.083, 0.404, 0.425, 0.487, 0.764, 0.829 spect, spect conv, abs 0.829 (5%) 0.8 abs
$^{212}_{82}\text{Pb}$ (ThB)		β^-	10.6 h	0.355, 0.589 spect, $\beta\gamma$ coinc 0.331, 0.569 (~12%) spect 0.340 spect 0.36 spect	γ_2 0.2386 spect conv γ_1 0.115, γ_2 0.176, γ_3 0.238, γ_4 0.249, γ_5 0.299 spect conv γ_3 0.238 (~40%), γ_5 0.300 (~4%) spect γ_3 (L_1/L_{II} ~18, M_1/M_{II} ~4.3) spect conv γ_3 (e/γ ~1), γ_5 (e/γ ~0.3) γ_3 0.238 spect
$^{214}_{82}\text{Pb}$ (RaB)		β^-	26.8 m	0.65 spect 0.72 spect	γ_1 0.05323, γ_2 0.24192, γ_4 0.29522, γ_5 0.35199 (γ_2/γ_4 / $\gamma_5 \approx 0.2/0.55$ / 1.0) cryst spect γ_1 0.0528, γ_2 0.2410, γ_3 0.2578, γ_4 0.2942, γ_5 0.3509 spect γ_1 (1.6%) crit abs γ_2 (K/L_1 ~6.7), γ_4 (K/L_1 ~6.7), γ_5 (K/L_1 ~5.9) spect conv 0.053, 0.241, 0.257, 0.294, 0.350 spect 0.053, 0.242, 0.296, 0.351 spect conv
$^{218}_{83}\text{Bi}$		α	1.7 m	6.2 ion ch	
$^{198}_{83}\text{Bi}$		EC 99+%, α $5 \times 10^{-3}\%$	7 m genet	5.83 ion ch	
$^{199}_{83}\text{Bi}$		EC 99+%, α $10^{-3}\%$	~25 m genet	5.47 ion ch, abs mica	
$^{200}_{83}\text{Bi}$		EC	35 m genet		
$^{201}_{83}\text{Bi}$		EC 99+%, α $3 \times 10^{-3}\%$	62 m	5.15 ion ch	
$^{201}_{83}\text{Bi}$		EC	~2 h genet		
$^{202}_{83}\text{Bi}$		EC	95 m		
$^{203}_{83}\text{Bi}$		EC $\alpha \sim 10^{-6}\%$	12 h genet	α : 4.85 range emuls	

Isotope Z A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
				Particles	Gamma transitions
$_{83}\text{Bi}^{204}$		EC, no β^+	12 h	Conv: ~ 0.2 , ~ 0.8 (weak), abs, spect	0.217 spect conv
Bi^{206}		EC	14.5 d		0.431, 0.527, 0.550, 0.746, 1.84 spect conv
Bi^{208}		EC EC, no β^+	6.4 d		0.182, 0.234, 0.260, 0.341, 0.396, 0.470, 0.505, 0.536, 0.590, 0.803, 0.880, 0.889, 1.020, 1.097, 1.720 spect, spect conv, γ - γ coinc Others
Bi^{207}		EC	~ 50 y genet		0.56, 1.1 (coinc with 0.56 γ) scint spect, γ - γ coinc 0.064 or 0.137, 0.565, 1.063, 1.46, 2.06, 2.20, 2.33, 2.49 spect conv With $\text{Pb}^{207\text{m}}$: 0.5, 1.1 scint spect
Bi^{208}		EC	Long		No γ (?)
Bi^{209}	100				
Bi^{209}		α	$> 10^{17}$ y sp act 3×10^{17} y sp act	~ 3.15 range emuls	
Bi^{210} (RaE)		β^- 99+%, α $5 \times 10^{-6}\%$	5.02 d 4.85 d 5.0 d 5.1 d	β^- : 1.17 spect	No γ
Bi^{210}		α , β^- , or IT ($\sim 0.3\%$)	$\sim 10^6$ y yield	4.93 ion ch	
Bi^{211} (AcC)		α 99.68%, β^- 0.32%	2.16 m	α : 6.618 (84%), 6.272 (16%) spect 6.621 (82.6%), 6.274 (17.4%) spect	0.350 abs ~ 0.35 ($e_K/\gamma \approx 0.18$, K/L 5.5) 0.354 abs
Bi^{212} (ThC)		β^- 66.3%, α 33.7%	60.5 m	α : 6.086 (27.2%), 6.047 (69.9%), 5.765 (1.7%), 5.622 (0.15%), 5.603 (1.1%), 5.481 (0.016%) spect 6.082 (27%), 6.042 (70%), 5.760 (1.8%), 5.618 (0.2%), 5.599 (1.1%) spect	With α : 0.040 (strong), 0.144, 0.164, 0.288, 0.328, 0.432, 0.452, 0.472, other γ 's, spect, spect conv 0.040 ($\sim 4\%$) coinc abs (e/γ ≥ 14) 0.295 spect conv

Isotope Z A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
				Particles	Gamma transitions
⁸³ Bi ²¹² (ThC) (Cont.)				β^- : 2.250 spect 2.256 spect, $\beta\gamma$ coine	With β^- : 2.20, 1.81 (~7%), 1.61 (~7%), 1.34 (~5%), 1.03 (~6%), 0.83 (~19%), 0.72 (~19%) spect 0.726 (~6%, e/γ 0.015) spect
⁸³ Bi ²¹³		β^- 98%, α 2%	47 m 46 m	β^- : 1.39 (68%), 0.959 (32%) spect ~1.3 abs α : 5.86 ion ch 6.0 ion ch	0.434 spect conv, scint spect
⁸³ Bi ²¹⁴ (RaC)		β^- 99+%, α 0.04%	19.7 m	α : 5.505 (45%), 5.444 (55%) spect 5.52 (37%), 5.47 (46%), 5.33 (17%) spect β^- : 3.17 (~23%), 1.65 (~77%) spect 3.15 spect, abs	With β^- : γ_1 0.6094 cryst spect γ_1 0.606 (K/L 5.6), γ_2 0.763, γ_3 0.933, γ_4 1.120 (K/L 6.7), γ_5 1.238 (K/L 5.9), γ_6 1.379, γ_7 1.520, γ_8 1.761 (K/L 6.7), γ_9 1.820, γ_{10} 2.200, γ_{11} 2.420 (γ_1/γ_2 / $\gamma_3/\gamma_4/\gamma_5/\gamma_6$ / $\gamma_7/\gamma_8/\gamma_9/\gamma_{10}$ / $\gamma_{11} \approx 9/1.3$ / 1.1/2.6/1.0/0.9/ 0.7/3.2/0.2/ 1.00/0.5) spect, spect conv 0.426, 0.498, 0.807, 0.766, 0.933, 1.120, 1.238, 1.379, 1.414, 1.761, 2.193 spect conv 0.609, 0.769, 0.935, 1.122, 1.241, 1.419, 1.766 spect conv With α : 0.0625, 0.191 spect conv
⁸³ Bi ²¹⁶		β^-	8 m		
⁸⁴ Po ²⁰⁰		EC, α	11 m	5.84 ion ch	
Po ²⁰¹		EC, α	18 m	5.70 ion ch	
Po ²⁰³		EC, α	52 m genet	5.59 ion ch	

Isotope Z A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
				Particles	Gamma transitions
^{203}Po		EC	47 m genet		
Po^{204}		EC ~99%, α ~1%	3.8 h	5.37 ion ch	
Po^{205}		EC 99+%, α 0.074%	1.5 h genet	5.2 ion ch	
Po^{206}		EC ~90%, α ~10%	9 d	5.218, 5.064 spect 5.21 ion ch	0.8 abs
Po^{207}		EC 99+%, α ~ $10^{-3}\%$	5.7 h	5.10 ion ch	1.3 abs
Po^{208}		α	2.93 y	5.108 spect 5.109 spect 5.10 ion ch	No γ
Po^{209}		α $\geq 90\%$, EC $\leq 10\%$ est	~100 y yield	4.877 spect 4.86 ion ch	0.87 (1%), 0.55 (0.5%), 0.2 (0.2%), 0.1 (0.07%) scint spect
Po^{210}		α β stable (cons energy)	138.3 d 140 d	5.298 spect 5.303 spect ~4.5 (weak) α - γ coinc, scint spect	γ_1 0.800 (e/γ ~0.03, K/L 3.7) spect, spect conv γ_1 0.804 scint spect γ_1 0.784 spect conv γ_1 0.773 spect γ_1 ($1.8 \times 10^{-3}\%$, e/γ ~0.07), no 0.08 γ ion ch, crit abs, γ - γ coinc γ_1 ($1.6 \times 10^{-3}\%$) abs 0.77 (~ $10^{-3}\%$), 0.084 (~ $10^{-3}\%$) abs Others
Po^{211} (AcC')		α β stable (cons energy)	0.52 s	7.434 spect 6.88 (0.50%), 6.56 (0.53%), no 6.34 α (lim 0.02%) spect 6.90 (0.6%), 6.57 (0.5%), 6.34 (0.1%) ion ch	
Po^{211}		α	25 s	7.14 ion ch	
Po^{212} (ThC')		α	3.04×10^{-7} s delay coinc 3.0×10^{-7} s delay coinc	8.776 spect Long range α 's: 10.536 (0.017%), 10.417 (0.002%), 9.489 (0.004%) spect	

Isotope Z A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
				Particles	Gamma transitions
$^{213}_{84}\text{Po}$		α	4.2×10^{-8} s delay coin	8.336 ion ch 8.34 ion ch	
$^{214}_{82}\text{Po}$ (RaC')		α β stable (cons energy)	1.637×10^{-4} s delay coin $\sim 1.5 \times 10^{-4}$ s delay coin 1.4×10^{-4} s delay coin	7.680 spect 7.683 spect Long range α 's: 9.069 (0.002%), others 8.280-10.509 spect	
$^{215}_{82}\text{Po}$ (AcA)		α 99+%, β^- $5 \times 10^{-4}\%$	1.83×10^{-3} s delay coin	7.365 spect	
$^{216}_{82}\text{Po}$ (ThA)		α β stable (cons energy) α 99+%, β^- 0.014%	0.158 s	α : 6.774 spect	
$^{217}_{82}\text{Po}$		α		6.5 ion ch	
$^{218}_{82}\text{Po}$ (RaA)		α 99+%, β^- 0.03%	3.05 m	α : 5.998 spect	
$^{219}_{85}\text{At}$		α , EC	43 s	6.50 ion ch	
$^{219}_{85}\text{At}$		α , EC	1.7 m	6.35 ion ch	
$^{220}_{85}\text{At}$		α , EC	7 m	6.10 ion ch	
$^{220}_{85}\text{At}$		EC	~ 25 m genet		
$^{220}_{85}\text{At}$		α , EC	25 m	5.90 ion ch	
$^{220}_{85}\text{At}$		EC	2.6 h genet		
$^{220}_{85}\text{At}$		EC 99+%, α 0.5%	1.7 h	5.65 ion ch	
$^{220}_{85}\text{At}$		EC $\sim 95\%$, α $\sim 5\%$	5.5 h	5.65 ion ch	0.2, 0.8 scint spect
$^{210}_{85}\text{At}$		EC 99+%, α 0.17% EC 99+%, α 0.1%	8.3 h	5.519 (32%), 5.437 (31%), 5.355 (37%) spect	0.25, 1.15, 1.40 scint spect 1.0 abs, abs conv
$^{211}_{85}\text{At}$		α 40.9%, EC 59.1%	7.5 h 7.3 h	5.862 spect 5.89 ion ch	
$^{212}_{85}\text{At}$		α	0.25 s		
$^{212}_{85}\text{At}$		α		9.2 range emuls	
$^{214}_{85}\text{At}$		α	$\sim 2 \times 10^{-8}$ s est	8.78 ion ch	

Isotope Z A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
				Particles	Gamma transitions
⁸⁵ At ²¹⁵		α	$\sim 10^{-4}$ s delay coinc Short	8.00 ion ch 8.4 ion ch	
At ²¹⁶		α	$\sim 3 \times 10^{-4}$ s delay coinc Short (<54 s)	7.79 ion ch 7.64 ion ch	
At ²¹⁷		α	0.018 s delay coinc 0.021 s delay coinc	7.02 ion ch 7.00 ion ch	
At ²¹⁸		α α 99+%, β^- 0.1%	1.5-2.0 s Several sec (?)	α : 6.63 range 6.7 ion ch	
At ²¹⁹		α $\sim$ 97%, β^- $\sim$ 3%	0.9 m	α : 6.27 ion ch	
⁸⁶ Em ²⁰⁸		EC $\sim$ 80%, α $\sim$ 20%	23 m	6.138 spect	
Em ²⁰⁹		EC $\sim$ 80%, α $\sim$ 20%	30 m	6.02 ion ch	
Em ²¹⁰		α >95%, EC <5%	2.7 h 2.1 h	6.036 spect 6.02 ion ch	
Em ²¹¹		EC 75%, α 25%	16 h	5.847 (33%), 5.778 (67%) spect 5.82 ion ch	
Em ²¹²		α	23 m	6.262 spect 6.23 ion ch	
Em ²¹⁵		α	$\sim 10^{-6}$ s est	8.6 ion ch	
Em ²¹⁶		α β stable (cons energy)	$\sim 10^{-4}$ s est	8.01 ion ch	
Em ²¹⁷		α β stable (cons energy)	$\sim 10^{-3}$ s delay coinc	7.74 ion ch	
Em ²¹⁸		α β stable (cons energy)	0.019 s delay coinc	7.12 ion ch	
Em ²¹⁹ (An)		α	3.92 s	6.824, 6.559, 6.434 spect α_0 (69%), α_{270} (15%), α_{337} (12%), α_{632} (4%) spect	0.067, 0.124, 0.198 (strong), 0.267, 0.321, 0.392, 0.589 (weak) spect conv 0.123, 0.270, 0.590 cryst spect
Em ²²⁰ (Tn)		α β stable (cons energy)	54.5 s	6.282 spect	

Isotope Z A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
				Particles	Gamma transitions
$^{221}_{86}\text{Em}$		β^- ~80%, α ~20%	25 m		
$^{223}_{86}\text{Em}$ (Rn)		α No β^- (lim $10^{-4}\%$)	3.825 d	5.486 spect	
$^{211}_{87}\text{Fr}$		EC 56%, α 44%	19.3 m	6.409 (37%), 6.387 (39%), 6.339 (24%) spect 6.36 ion ch	
$^{217}_{87}\text{Fr}$		α		8.3 range emuls	
$^{218}_{87}\text{Fr}$		α	5×10^{-3} s est	7.85 ion ch	
$^{219}_{87}\text{Fr}$		α β stable (cons energy)	0.02 s delay coinc	7.30 ion ch	
$^{220}_{87}\text{Fr}$		α	27.5 s	6.69 ion ch	
$^{221}_{87}\text{Fr}$		α	4.8 m 5 m	6.30 ion ch 6.30 (~75%), 6.05 (~25%) ion ch	0.220 spect conv, scint spect
$^{222}_{87}\text{Fr}$		β^- 99+%, α 0.01-0.1%	14.8 m		
$^{223}_{87}\text{Fr}$ (AcK)		β^- α $4 \times 10^{-3}\%$	21 m	β^- : 1.2 cl ch	0.09 abs ~0.330 (6%) abs, 0.0486 (27%) crit abs
$^{213}_{88}\text{Ra}$		α	2.7 m	6.90 ion ch	
$^{219}_{88}\text{Ra}$		α	$\sim 10^{-3}$ s est	8.0 ion ch	
$^{220}_{88}\text{Ra}$		α	3×10^{-3} s est	7.43 ion ch	
$^{221}_{88}\text{Ra}$		α	30 s	6.71 ion ch	
$^{222}_{88}\text{Ra}$		α β stable (cons energy)	38 s	6.51 ion ch	
$^{223}_{88}\text{Ra}$ (AcX)		α β stable (cons energy)	11.2 d	5.860 (weak), 5.730 (9%), 5.704 (53%), 5.596 (24%), 5.528 (9%), 5.487 (2%), 5.419 (3%) spect 5.750 (11%), 5.719 (53%), 5.607 (25%), 5.540 (9%), 5.433 (2%) spect No 6.0-6.1 α (lim 0.5%)	0.144, 0.155, 0.180, 0.270, 0.340 cryst spect 0.026, 0.064, 0.081, 0.099, 0.116, 0.154, 0.164, 0.180, 0.232, 0.268, 0.280, 0.322, 0.348, 0.444 spect conv
$^{224}_{88}\text{Ra}$ (ThX)		α β stable (cons energy)	3.64 d	5.681 (95%), 5.448 (4.6%), 5.194 (0.4%) spect 5.681 spect 5.66 ion ch	0.241 (e_K/γ 0.1) spect conv No γ

Isotope Z A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
				Particles	Gamma transitions
^{226}Ra		β^- No α (lim 0.1%)	14.8 d 14 d	~ 0.31 spect ~ 0.2 abs < 0.05 abs	
^{226}Ra		α β stable (cons energy)	1,622 y sp act 1,631 y sp act 1,590 y sp act	α_0 4.777 spect α_0 (94.3%), α_{188} (5.7%) spect α_0 (95.2%), α_{188} (4.8%) ion ch α_0 (93.5%), α_{187} (6.5%) spect	0.186 spect conv ~ 0.19 (e/γ 0.88, $K/L \sim 0.5$) α -conv coinc abs 0.188 ($e/\gamma \sim 0.5$)
^{227}Ra		β^-	41.2 m	1.30 spect	0.291, 0.498 scint spect
^{228}Ra (MsTh_1)		β^-	6.7 y	~ 0.012 (?) cl ch 0.053 spect, abs	~ 0.03 cl ch
^{229}Ra		$[\beta^-]$	[Short]		
^{230}Ra		β^-	1 h	1.2 abs, spect	
^{221}Ac		α		7.6 range emuls	
^{222}Ac		α	5.5 s	6.96 ion ch	
^{223}Ac		α 99%, EC 1%	2.2 m	6.64 ion ch	
^{224}Ac		$EC \sim 90\%$, $\alpha \sim 10\%$	2.9 h	6.17 ion ch	
^{225}Ac		α β stable (cons energy)	10.0 d	5.80 ion ch	
^{226}Ac		β^-	29 h	1.17 abs	
^{227}Ac		$\beta^- \sim 99\%$ α 1.2%	22.0 y 21.7 y 13.5 y	α : 4.942 spect 4.94 (100%) ion ch Others β^- : 0.04 spect ~ 0.02 cl ch < 0.03 abs	0.037 (weak) abs (0.2%)
^{228}Ac (MsTh_2)		β^-	6.13 h	2.18 (10%), 1.85 (9%), 1.72 (7%), 1.15 (53%), 0.66 (8%), 0.46 (13%) spect 2.03, 1.74, 1.10 spect Others	0.058, 0.129, 0.184, 0.338, 0.462, 0.914, 0.969 spect conv 0.333, 0.462, 0.913, 0.968 spect 0.063, 0.146, 0.186, 0.338, 0.533, 0.590, 0.905 spect conv

Isotope Z A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
				Particles	Gamma transitions
$^{229}_{89}\text{Ac}$		β^-	66 m		
Ac^{230}		β^-	<1 m genet	~2.2 abs, spect	
$^{223}_{90}\text{Th}$		α	~0.1 s est	7.55 ion ch	
Th^{224}		α β stable (cons energy)	~1 s est	7.13 ion ch	
Th^{225}		α ~90%. EC ~10%	8.0 m	6.57 ion ch	
Th^{226}		α β stable (cons energy)	30.9 m	α_0 (78%), α_{117} (22%) spect 6.30 ion ch	
Th^{227} (RdAc)		α β stable (cons energy)	18.6 d 18.9 d	6.030 (19%), 6.001 (5%), 5.972 (21%), 5.952 (13%), 5.922 (~2%), 5.860 (4%), 5.796 (2%), 5.749 (17%), 5.728 (~1%), 5.704 (15%), 5.651 (~2%) spect 6.049 (20%), 6.017 (4%), 5.988 (25%), 5.966 (4%), 5.922 (~1%), 5.868 (2.5%), 5.815 (~1%), 5.764 (20%), 5.742 (4%), 5.717 (15%), 5.672 (2.5%) spect	0.050, 0.057, 0.080, 0.101, 0.113, 0.129, 0.208, 0.240, 0.258 cryst spect 0.050 (3%), 0.120 (13%), 0.280 (50%) abs 0.050 (5%), 0.125 (10%), 0.270 (26%) abs 0.050 (~3%), 0.125 (23%) abs
Th^{228} (RdTh)		α β stable (cons energy)	1.90 y	5.423 (72%), 5.338 (28%) spect	0.0843 spect conv 0.083 (e/ γ 12) α - conv coinc abs 0.083 (2.1%, e/ γ $\approx$ 10) crit abs
Th^{229}		α β stable (cons energy)	7,340 y genet ~ 10^4 y genet	5.02 (~10%), 4.94 (~20%), 4.85 (~70%) ion ch	
Th^{230} (Io)		α β stable (cons energy)	8.0×10^4 y sp act 8.2×10^4 y genet	4.682 (~75%), 4.613 (~25%), 4.51 (?) (weak) spect 4.66 range air ion ch	0.068, 0.228 (very weak) spect conv 0.068 (0.85%), 0.140 (0.33%), 0.240 (0.05%) abs ~0.07 (coinc with α , e/ γ ~46) α -conv coinc abs ~0.068 (0.5%), 0.190 (0.3%) abs ~0.07 (coinc with α), ~0.2 (coinc with α) α - γ , α - conv coinc abs

Isotope Z A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
				Particles	Gamma transitions
^{231}Th (UY)		β^-	25.64 h	0.302 (44%), 0.216	γ_1 0.022, γ_2 0.059,
			25.5 h	(11%), 0.094 (45%)	γ_3 0.063, γ_4 0.085,
			24.6 h	spect	γ_5 0.107, γ_6 0.122,
			24.0 h	0.39 (~20%), 0.19 (~40%), 0.10 (~40%) spect 0.2 abs	γ_7 0.167, γ_8 0.208, γ_9 0.230 ($\gamma_2 +$ $\gamma_3/\gamma_4/\gamma_5/\gamma_6/\gamma_7/$ $\gamma_8/\gamma_9 \approx 0.40/$ 1.00/0.065/0.02/ 0.018/0.003/ 0.001) spect conv, scint spect 0.059, 0.063, 0.082, 0.120 spect conv 0.035 (>80%), 0.210 abs
^{232}Th	100	α β stable (cons energy)	1.39×10^{10} y	3.98 ion ch	~0.055 (coinc with
			sp act Spont fission: 1.4×10^{14} y	3.98 range emuls 4.20 ion ch	24% of α) α -conv coinc emuls ~0.075 (coinc with ~20% of α) α - conv coinc emuls
^{233}Th		β^-	23.3 m	1.23 spect	0.098 (0.25%), 0.172
			23.5 m	1.24 spect	(0.03%), 0.350
			23.6 m	1.2 abs	(0.004%), 0.448
			23 m		(0.1%), 0.662 (0.05%) scint spect No γ
^{234}Th (UX ₁)		β^-	24.10 d	0.205 (~80%), 0.111	0.090 (e/γ ~0.2)
			24.1 d	(~20%) spect	spect conv
			24.5 d	0.192 (56%), 0.104	0.093 (~20%, e_L/γ
				(44%) spect 0.190 spect 0.20, 0.1 abs	~0.34), 0.180 (4.5%) spect conv, abs 0.092
^{235}Th		$[\beta^-]$	<10 m genet		
^{235}Pa		α	2.0 s		
^{236}Pa		α	1.8 m	6.81 ion ch	
^{237}Pa		α ~85%, EC ~15%	38.3 m	6.46 ion ch	
^{238}Pa		EC ~98%, α ~2%	22 h	6.09 (75%), 5.85 (25%) ion ch	
^{239}Pa		EC 99+%, α 0.25%, EC ~99%, α ~1%	1.5 d	5.69 ion ch 5.66 ion ch	
^{230}Pa		EC ~92%, β^- ~8%, α ~0.003%	17.7 d 17.0 d	β^- : ~0.43 abs	0.94 abs

Isotope Z A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
				Particles	Gamma transitions
^{231}Pa	α β stable (cons energy)	3.43×10^4 y sp act 3.2×10^4 y sp act		5.042 (11%), 5.002 (47%), 4.938 (25%), 4.838 (3%), 4.720 (11%), 4.660 (1-3%) spect ~ 5.0 (85%), ~ 4.7 (15%) ion ch	0.034, 0.038, 0.057, 0.064, 0.082, 0.102, 0.198, 0.259, 0.301, 0.331, 0.357, 0.383 spect conv 0.095, 0.294, 0.323 spect conv 0.027 (9%), 0.087, 0.100, 0.30 (4%) abs, crit abs 0.044, 0.066 cl ch 0.027 crit abs, ion ch Others
^{232}Pa	β^- No EC (lim 2%)	1.32 d 1.4 d 1.6 d		0.99 ($\sim 30\%$), 0.64 ($\sim 10\%$), 0.28 ($\sim 60\%$) spect ~ 0.28 abs	~ 0.23 ($\sim 30\%$), 1.05 ($\sim 100\%$) abs 0.21, 1.0 abs
^{233}Pa	β^-	27.4 d		0.530, 0.430 spect 0.5 abs ~ 0.2 spect, abs ~ 0.7 spect	0.029, 0.041, 0.058, 0.076, 0.087, 0.105, 0.273, 0.302, 0.313, 0.342, 0.377, 0.400, 0.416 spect conv 0.028, 0.040, 0.058, 0.076, 0.087, 0.105, 0.302, 0.343, 0.417 spect conv 0.0287, 0.0405, 0.0754, 0.0870 (rel abund 100/ 75/3/3) cryst spect 0.084, 0.298, 0.309, 0.337 spect conv No $\gamma > \sim 0.4$ abs
^{234}Pa (UX ₂)	β^- 99+%, IT 0.15%	1.175 m 1.14 m		2.32 (80%), 1.50 (13%), 0.60 (?) ($\sim 7\%$) spect 2.32 (98%) spect	0.817 (e_K/γ 0.04) spect conv 0.394 (with IT, 0.15%, $e/\gamma \sim 1$), ~ 0.9 ($\sim 2\%$), ~ 1.5 ($\sim 0.2\%$) spect, spect conv 0.822, 0.806, 0.782 (weak, e/γ large) spect conv
^{234}Pa (UZ)	β^-	6.7 h		~ 1.2 (10%), 0.45 (90%) spect 1.6, 0.6 abs	0.85 (two γ 's) abs, β - γ coine ~ 0.7 (two γ 's) abs, γ - γ coine
^{235}Pa	β^-	23.7 m 23 m		1.4 abs	No γ , abs

Isotope Z A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
				Particles	Gamma transitions
$^{227}_{92}\text{U}$		α	1.3 m	6.8 ion ch	
$^{228}_{92}\text{U}$		$\alpha \sim 80\%$, $EC \sim 20\%$	9.3 m	6.67 ion ch	
$^{229}_{92}\text{U}$		$EC \sim 80\%$, $\alpha \sim 20\%$	58 m	6.42 ion ch	
$^{230}_{92}\text{U}$		α β stable (cons energy)	20.8 d	5.85 ion ch α_0 (77%), α_{70} (23%) spect	
$^{231}_{92}\text{U}$		EC 99+%, α $5.5 \times 10^{-4}\%$	4.3 d 4.2 d	5.45 ion ch	0.051, 0.064, 0.076 spect conv
$^{232}_{92}\text{U}$		α β stable (cons energy)	70 y yield ~ 30 y yield	5.31 range Al 5.27 range air α_0 (69%), α_{58} (31%) spect	~ 0.060 (coinc with 30% of α) α -conv coinc emuls
$^{233}_{92}\text{U}$		α β stable (cons energy)	1.62×10^5 y sp act + mass spect 1.63×10^5 y sp act + mass spect 1.2×10^5 y yield	4.823 ion ch α_0 (83%), α_{44} (15%), α_{94} (2%) spect 4.80 abs air Others	0.0428 (0.05%), 0.0561 (0.01%) ion ch ~ 0.043 , ~ 0.056 , 0.099 α -conv coinc emuls 0.04, 0.08 (0.8%, e/γ ~ 8), 0.31 (0.1%, $e/\gamma \sim 3$) abs 0.04, 0.09, 0.36 (all coinc with α) α - γ , α -conv coinc abs
$^{234}_{92}\text{U}$ (U_{11})	0.0058	α β stable (cons energy)	2.48×10^5 y sp act 2.52×10^5 y sp act, sp act + mass spect 2.67×10^5 y yield 2.35×10^5 y sp act + mass spect Spont fission: 2×10^{16} y	4.763 ion ch α_0 (74%), α_{47} (26%) spect 4.76 ion ch 4.78 range air, ion ch	0.050, 0.117 scint spect γ_1 0.053, γ_2 0.093, γ_3 0.118 (γ_1/γ_2 / $\gamma_3 = 1/\sim 0.2$ / 0.4) scint spect 0.055 (coinc with α , e/γ large) α -conv coinc emuls 0.065 (coinc with α , e/γ large) α - γ coinc abs, α -conv coinc abs Others
$^{235}_{92}\text{U}$ (AcU)	0.715	α β stable (cons energy)	7.13×10^8 y sp act 7.07×10^8 y radiogenic Pb ratios Spont fission: 1.9×10^{17} y	4.58 (10%), 4.47 (?) ($\sim 3\%$), 4.40 (83%), 4.20 (4%) ion ch 4.39 ion ch	γ_1 0.094, γ_2 0.143, γ_3 0.184, γ_4 0.289, γ_5 0.386 (γ_1/γ_2 / $\gamma_3/\gamma_4/\gamma_5 = 0.9$ / 0.2/1.0/0.1/0.05) scint spect 0.187 abs 0.167 abs
$^{238}_{92}\text{U}$		α β stable (cons energy)	2.39×10^7 y sp act 2.46×10^7 y sp act	4.499 ion ch 4.5 ion ch	~ 0.050 (coinc with 27% of α) α -conv coinc emuls

Isotope Z	A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
					Particles	Gamma transitions
$_{92}\text{U}^{237}$			β^-	6.75 d 6.63 d 6.8 d	0.245, other β^- (weak) spect ~ 0.23 spect	γ_1 0.027, γ_2 0.043, γ_3 0.059, γ_4 0.165, γ_5 0.207 (eK/γ 1.6, K/L 5.0), γ_6 0.269, γ_7 0.334, γ_8 0.370, γ_9 0.430 ($\gamma_3/\gamma_5/\gamma_7 \approx 37/$ 21/2.5) scint spect, spect conv 0.0598 cryst spect 0.032, 0.057, 0.204, 0.260 abs, spect conv 0.14, 0.23, 0.53 abs
U^{238} (U_1)	99.28		α β stable (cons energy)	4.49×10^9 y sp act 4.51×10^9 y sp act Spont fission: 8.0×10^{16} y	4.18 ion ch 4.21 range	~ 0.045 (coinc with 22% of α) α -conv coinc emuls 0.048 (coinc with 23% of α) α -conv coinc emuls ~ 0.050 (coinc with 24% of α) α -conv coinc emuls
U^{239}			β^-	23.54 m 23.5 m	1.21 spect 1.20 abs 1.2 abs 1.12 spect	0.0736 spect conv, scint spect 0.073 spect conv, abs 0.076, >0.3 ($\leq 10\%$) abs
U^{240}			β^-	~ 18 h genet		
$_{93}\text{Np}^{231}$			α	~ 50 m	6.28 ion ch	
Np^{232}			EC	~ 13 m		Hard γ
Np^{233}			EC 99+%, $\alpha \sim 10^{-3}\%$	35 m	5.53 ion ch Conv: ~ 0.3	
Np^{234}			EC ($L/K \sim 1$) No α (lim 0.01%) No β^+	4.40 d 4.4 d		0.177, 0.442, 0.803, 1.42 spect conv 1.9 abs
Np^{235}			EC ($L/K > 9$) $\alpha \sim 5 \times 10^{-3}\%$	410 d	5.06 ion ch	
Np^{236}			$\beta^- \sim 33\%$, EC $\sim 67\%$ ($L/K \sim 2$)	22 h	0.51 ($\sim 60\%$), 0.36 ($\sim 40\%$) spect	0.150 (e/γ large) spect conv
Np^{237m}			IT	6.3×10^{-8} s delay coinc		0.060 scint spect
Np^{237}			α β stable (cons energy)	2.20×10^6 y sp act	4.77 ion ch	Soft γ 's (coinc with 80% of α) α -conv coinc emuls Soft γ , spect conv

Isotope Z A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
				Particles	Gamma transitions
^{238}Np		β^-	2.10 d 2.0 d	1.272 (47%), 0.258 (53%) spect, spect coinc 1.39, 0.22 abs	0.043, 0.047, 0.103 (L/M 1.6), 0.983, 1.03 (1.03 γ /0.98 γ $\approx$ 1) spect, spect conv, β - γ , β - x coinc No 0.047 γ 1.1 abs 0.075, 1.2 abs, abs conv
^{239}Np		β^-	2.33 d 2.35 d 2.3 d	0.715, 0.654, 0.44, 0.33 spect 0.718 (4.8%), 0.655 (1.7%), 0.441 (31%), 0.390 (10%), 0.329 (52%) spect 0.705 (7%, not coinc with γ), 0.435 (46%), 0.310 (47%) spect 1.179, 0.676, 0.403, 0.288 spect Others	0.049, 0.057, 0.061, 0.067 (coinc with 0.210 γ), 0.210, 0.227, 0.276 (last 3 γ 's coinc with 0.435 β^- , 0.276 γ not coinc with 0.227 γ) spect conv, β - γ , γ - γ coinc 0.013, 0.019, 0.044, 0.049, 0.057, 0.061, 0.067, 0.077, 0.105, 0.209, 0.228, 0.254, 0.277, 0.285, 0.316, 0.334 spect conv 0.044, 0.049, 0.057, 0.061, 0.068, 0.105, 0.209, 0.228, 0.254, 0.277, 0.286 spect conv 0.023, 0.049, 0.057, 0.061, 0.068, 0.106, 0.209, 0.228, 0.277 spect conv 0.0576, 0.0618 cryst spect 0.057, 0.061, 0.067, 0.206, 0.227, 0.275 spect conv Others
^{240}Np		β^-	7.3 m	$\sim$ 1.3 abs	
^{241}Np		β^-	60 m	0.89 spect	0.15, 0.20, 0.26, 0.58 spect conv $\sim$ 0.7 abs
^{232}Pu		$\alpha \geq 2\%$, $EC \leq 98\%$	36 m	6.58 ion ch	
^{234}Pu		$\alpha \sim 4\%$, $EC \sim 96\%$	9.0 h 8.5 h	6.19 ion ch 6.2 ion ch	
^{236}Pu		EC 99 + %, $\alpha \sim 0.002\%$	26 m	5.85 ion ch	

Isotope Z A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
				Particles	Gamma transitions
⁹⁴ Pu ²³⁶		α β stable (cons energy)	2.7 y Spont fission: 3.5×10^9 y	5.75 range air 5.75 ion ch	~ 0.045 (coinc with 20% of α) α -conv coinc emuls
Pu ²³⁷		EC	~ 40 d		K, L-x, no γ
Pu ²³⁸		α β stable (cons energy)	89.6 y 92 y genet 77 y genet Spont fission: 3.8×10^{10} y	5.492 (76%) 5.450 (24%) spect 5.47 range air 5.51 range air	0.045 spect conv ~ 0.040 (coinc with 23% of α) α -conv coinc emuls ~ 0.04 ($e/\gamma \sim 700$) α - γ coinc, scint spect
Pu ^{239m}		IT	1.1×10^{-9} s delay coinc		0.049, 0.067, 0.210, 0.227, 0.276 spect conv, coinc
Pu ²³⁹		α β stable (cons energy)	24,360 y sp act 24,400 y sp act 24,300 y sp act 24,100 y calorimetric Spont fission: 5.5×10^{15} y	5.150 (69%) 5.137 (20%) 5.099 (11%) spect 5.147 ($\sim 70\%$) 5.097 ($\sim 30\%$) spect 5.14 ion ch 5.15 range air 5.13 ion ch 5.13 range air Conv: ~ 0.10 (weak) α - conv coinc emuls	γ_1 0.039, γ_2 0.0531, γ_3 0.100, γ_4 0.124, γ_5 0.384 (γ_1/γ_2 / $\gamma_3/\gamma_4/\gamma_5 \approx 0.4$ / 1.4/1.1/0.5/0.3) spect conv, scint spect 0.0385 (2×10^{-3} %), 0.0520 (7×10^{-3} %) ion ch ~ 0.035 , 0.05 α -conv coinc emuls 0.05 α -conv coinc emuls Others
Pu ²⁴⁰		α β stable (cons energy)	6,580 y genet 6,240 y sp act 6,760 y sp act	5.162 (76%) (24%) spect	0.0496 spect conv, scint spect
Pu ²⁴¹		β^- 99+%, $\alpha \sim 10^{-3}\%$	14 y genet	α : 4.91 ion ch β^- : 0.021 spect ~ 0.02 abs	γ_1 0.100, γ_2 0.145 ($\gamma_1/\gamma_2 \approx 5$) scint spect
Pu ²⁴²		α β stable (cons energy)	$\sim 5 \times 10^6$ y genet	4.88 ion ch	
Pu ²⁴³		β^-	4.98 h 5.0 h	0.39 spect ~ 0.45 abs	0.095, 0.12 spect conv
⁹⁵ Am ²³⁷		EC 99+%, α 0.005%	~ 1.3 h	6.01 ion ch	
Am ²³⁸		EC No α (lim $3 \times 10^{-4}\%$)	2.1 h	Conv	
Am ²³⁹		EC 99+%, α 0.003%	12 h	5.75 ion ch	0.3 (10%) scint spect 0.3 abs conv

Isotope <i>Z</i> <i>A</i>	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
				Particles	Gamma transitions
²⁴⁰ Am		EC No α (lim 0.2%)	50 h 53 h		No γ > 0.7
²⁴¹ Am		α β stable (cons energy)	470 y sp act 475 y sp act	5.546 (0.23%), 5.535 (0.34%), 5.503 (0.21%), 5.476 (84.2%), 5.433 (13.6%), 5.379 (1.4%) spect 5.48 ion ch 5.47 range air	γ_1 0.0597 (40%, e/γ ≤ 1.5 , with Np ^{237m}), γ_2 0.0263 (2.8%, $e/\gamma \leq 20$), γ_3 0.0209, γ_4 0.0173, γ_5 0.0135 ($\gamma_1/\gamma_2/\gamma_3/\gamma_4/\gamma_5$ $\approx 1/0.069/0.155/$ 0.56/0.32) ion ch, α - γ delay coinc γ_1 0.0598, γ_2 0.0264 ($\gamma_1/\gamma_2 \approx 1/0.30/$ 0.05/0.10) cryst spect 0.0590, 0.0414, 0.0264 spect conv, scint spect Others
^{242m} Am		β^- 60%, EC (L) 20%, IT 20%	16.01 h 15.7 h 16 h	0.628 spect 0.63 abs	0.035, 0.038, 0.053 spect conv
²⁴³ Am		β^- , EC, α (α/β^- 0.01)	~ 100 y genet	β^- : 0.593 spect ~ 0.5 abs	0.038, 0.053 spect conv
²⁴³ Am		α	$\sim 10^4$ y genet	5.267 ($\sim 90\%$), 5.226 ($\sim 10\%$) spect 5.27 ion ch	0.075 (coinc with 90% of α) α - γ coinc, scint spect 0.076 scint spect
²⁴⁴ Am		β^-	~ 25 m		
²⁴⁸ Cm		EC < 90%, α > 10%	2.5 h	6.50 ion ch	
²⁴⁹ Cm		EC $\sim 100\%$, no α (lim 0.1%)	~ 3 h		
²⁴⁰ Cm		α No EC (lim 0.5%)	26.8 d Spont fission: 7.9×10^5 y	6.25 ion ch 6.3 range air	
²⁴¹ Cm		EC 99 + %, α 0.2%	35 d	5.90 ion ch	
²⁴² Cm		α β stable (cons energy)	162.5 d 162 d Spont fission: 7.2×10^6 y	6.110 (73.7%), 6.066 (26.3%), 5.965 (0.035%) spect 6.118 ion ch 6.08 ion ch	0.043 spect conv ~ 0.04 ($e/\gamma \sim 600$) α - γ coinc, scint spect ~ 0.045 (coinc with 23% of α) α -conv coinc emuls ~ 0.04 (coinc with α) α -conv coinc

Isotope Z A	Per cent abundance	Type of decay	Half-life	Energy of radiation, Mev	
				Particles	Gamma transitions
$^{96}\text{Cm}^{243}$		α	~ 100 y genet	5.985 (6.4%), 5.777 (80.5%), 5.732 (13.1%) spect 5.89 (15%), 5.79 (85%) ion ch	0.226, 0.277 (both γ 's coincide with 5.777 α) scint spect
Cm^{244}		α β stable (cons energy)	19 y Spont fission: 1.4×10^7 y	5.798 (75%), 5.755 (25%) spect	
Cm^{245}		α	> 500 y	~ 5.6 ion ch	
$^{97}\text{Bk}^{243}$		EC 99 + %, $\alpha \sim 0.1\%$	4.6 h	6.72 (30%), 6.55 (53%), 6.20 (17%) ion ch	
Bk^{244}		EC	~ 5 h		
Bk^{245}		EC 99 + %, $\alpha \sim 0.1\%$	4.95 d	6.33 (18%), 6.15 (48%), 5.90 (34%) ion ch	
$^{98}\text{Cf}^{244}$		α , EC (?)	45 m	7.15 ion ch	
Cf^{246}		α β stable (cons energy)	35.7 h Spont fission: $\sim 2,000$ y	6.75 ion ch	

Section 7

INTERACTION OF RADIATION WITH MATTER **(Beta Rays, Gamma Rays, X-rays, and Neutrons)**

By

E. P. BLIZARD, M.A., *Director, Applied Nuclear Physics Division, Oak Ridge
National Laboratory, Oak Ridge, Tennessee.*

Beta Rays.....	7-2
Gamma Rays and X-rays.....	7-5
Neutrons.....	7-10

INTERACTION OF RADIATION WITH MATTER

E. P. Blizard

BETA RAYS

Bibliography: Segrè (ed.), "Experimental Nuclear Physics," Wiley, 1953. Evans, "The Atomic Nucleus," McGraw-Hill, 1955.

Beta rays are electrons which are produced in nuclear radioactive decay. Thus the interaction of electrons with matter will indicate the penetration of beta rays, provided, of course, that the spectrum of the latter is also known.

Interaction Processes

There are several processes by which high-velocity electrons can interact with the matter through which they pass, the most important being (1) bremsstrahlung, (2) elastic scattering, and (3) inelastic scattering. The relative probabilities for interaction by these three processes, which are described below, vary strongly with the electron energy and to much less extent with the nature of the matter.

Bremsstrahlung. An electron can lose energy by emitting electromagnetic radiation when its velocity is changed suddenly in magnitude or direction (i.e., when it is accelerated). High-energy X-rays are normally produced by this process, i.e., by the sudden stopping of high-energy electrons in a heavy-metal target. Since the greatest accelerations are accomplished on high-speed electrons traversing the coulomb fields of highly charged nuclei, bremsstrahlung is the most important interaction process for high-energy electrons in high- Z elements (Z is the atomic number). In lead it is the dominant process for electrons of energies >6.9 Mev, in copper for >21.5 Mev, in aluminum for >47 Mev, and in hydrogen for >340 Mev [Bethe and Ashkin, in Segrè, E. (ed.), "Experimental Nuclear Physics," vol. I, Wiley, New York, 1953]. The X-rays so produced are, of course, much more penetrating than the electrons that produced them and in general travel some distance before being stopped.

Elastic Scattering. An electron can be diverted elastically in the coulomb field of an atom. Since the atom is so much heavier than the electron, this results in essentially no transfer of energy, although significant change of direction can take place. The effect is important mainly in that it makes the total effective travel of the electron shorter by making the slowing-down path more tortuous.

Inelastic Scattering—Ionization and Excitation. The traveling electron can excite or ionize the atoms of the materials through which it passes by coulomb inter-

action. By this means energy is deposited in the medium along the path. This is the dominant process by which electrons are slowed down below the energy limits given under Bremsstrahlung.

Typical History of a High-energy Electron

A typical history of a **high-energy electron attenuation** would be: first, bremsstrahlung radiation emission, provided the electron starts with sufficiently high energy; next, many inelastic collisions causing heavy ionization along the path as well as excitation of atoms; next, at quite low energies, many elastic scatterings but not much travel; and, finally, capture in the lattice. Statistically, some slowed-down electrons will travel much farther than others before capture, giving rise to an observably greater penetration. This phenomenon is called **straggling**.

Total Effective Electron Travel, or "Range"

The total effective travel of an electron is characterized by a **range**, expressed in grams per square centimeter, which is the product of the density of material and the distance of travel. The latter is not well defined because of the statistical nature of the interactions, with large energy losses permissible in a single collision. However, in a plot of the number of electrons penetrating a slab of material vs. the thickness of penetration, there is a linear portion just above the turnup due to straggling. Extrapolation of this linear portion gives a reasonably well-defined range R which is a unique function of the original electron energy E . It is possible to fit the range-energy relation empirically by relatively simple formulas, as was first shown by Feather (*Proc. Cambridge Phil. Soc.*, vol. 34, p. 599, 1938). Marshall and Ward (*Can. J. Research*, vol. A15, p. 39, 1937) have developed a formula for the **range of monoenergetic electrons**, in grams per square centimeter of Aluminum, as follows:

$$R = 0.526E - 0.094 \text{ g/cm}^2 \quad E > 0.6 \text{ Mev}$$

For the case in which the electrons derive from **nuclear beta decay**, the extrapolation is somewhat different. In this case the extrapolated range is better given as a function of the maximum electron energy E_0 . Empirical formulas developed by several investigators are as follows:

1. Glendinin (*Nucleonics*, vol. 2, p. 12, 1948)

$$R = 0.542E_0 - 0.133 \text{ g/cm}^2 \quad E_0 > 0.8 \text{ Mev}$$

$$R = 0.407(E_0)^{1.38} \text{ g/cm}^2 \quad 0.15 \text{ Mev} < E_0 < 0.8 \text{ Mev}$$

2. Flammersfeld (*Nature*, vol. 33, p. 280, 1946)

$$R = 0.11(\sqrt{1 + 22.4E_0^2} - 1) \text{ g/cm}^2 \quad 0 < E_0 < 3 \text{ Mev}$$

3. Bleuler and Zunti (*Helv. Phys. Acta*, vol. 19, p. 137, 1946)

$$R = 0.571E_0 - 0.161 \text{ g/cm}^2 \quad E_0 > 1 \text{ Mev}$$

Note that the formulas best fit the data for aluminum but that they apply reasonably well to other materials, especially other light elements. Corrections to other

elements can be made theoretically, using the work of Bethe (*Ann. Physik*, vol. 5, p. 325, 1930; "Handbuch der Physik," vol. 24, p. 273, Springer, Berlin, 1953).

Beta-ray Energy Determination

It is possible, using the foregoing range formulas, to determine the **energy of beta rays from absorption data**. This is best done by comparison with the absorption data of a known standard emitter, according to a method devised by Feather (*Proc. Cambridge Phil. Soc.*, vol. 34, p. 599, 1938). Bleuler and Zunti (*Helv. Phys. Acta*, vol. 19, p. 383, 1946) have extended and improved the method, giving parameters of attenuation for a range of energies (beta-emission maximum) from 0.4 to 5.5 Mev, from which it is possible to identify quite accurately the energies of beta emitters from aluminum absorption data.

Rate of Energy Loss of Electrons

The rate of **energy loss of electrons** by ionization and excitation is useful in determining the response of beta and gamma-ray detectors, and can be calculated by the formula worked out by Bethe ("Handbuch der Physik," vol. 24, p. 273, Springer, Berlin, 1953):

$$-\frac{dE}{dx} = \frac{2\pi Ne^4 Z}{mv^2} \left[\log_e \frac{mv^2 E}{2I^2(1-\beta^2)} - (2\sqrt{1-\beta^2} - 1 + \beta^2) \log_e 2 + 1 - \beta^2 + \frac{1}{8}(1 - \sqrt{1-\beta^2})^2 \right] \quad (7-1)$$

where N = number of atoms per cubic centimeter of the material being traversed

Z = nuclear charge of the material

e = electronic charge

m = electronic mass

v = velocity of the electron, cm/sec

E = kinetic energy of the electron (total energy minus rest energy)

I = geometric mean excitation and ionization potential of the atoms of the material, ev

$I = 13.5Z$, or see a more complete treatment by Evans ("The Atomic Nucleus," p. 580, McGraw-Hill, 1955)

$\beta = v/c$

c = velocity of light

For nonrelativistic velocities, $\beta \ll 1$, this expression reduces to

$$-\frac{dE}{dx} = \frac{4\pi e^2 NZ}{mv^2} \log_e \frac{mv^2}{2I} \sqrt{\frac{e}{2}} \quad (7-2)$$

Ionization of a Gas

The average amount of energy spent in the production of one ion pair varies only slightly with the velocity or charge of the primary particle. Values collected

by Staub [Segrè, E. (ed.), "Experimental Nuclear Physics," vol. I, p. 3, Wiley, New York, 1953] for the **energy loss per ion pair** are given in Table 7-1.

Table 7-1. Electron Energy Loss per Ion Pair, I_0

Gas	Electron Energy, Mev	I_0 , ev
Air.....	3.0	32
Argon.....	17.4	26.9
Xenon.....	0.042	21.3

GAMMA RAYS AND X-RAYS

Bibliography: White, X-ray Attenuation Coefficients, *Natl. Bur. Standards (U.S.) Rept. 1003*, 1952. Davisson and Evans, *Revs. Modern Phys.*, vol. 24, p. 79, 1952. Fano, U., *Nucleonics*, vol. 11, no. 8, p. 8, 1953. U.S. Atomic Energy Commission, "Reactor Handbook," vol. 1, Sec. 2, McGraw-Hill, 1955. *AECD Rept. 3645*, 1955.

Gamma rays and **X-rays** are electromagnetic radiation of high energy per photon. There is no difference between the two radiations, but the term "gamma ray" implies radiation of nuclear origin whereas "X-ray" is reserved for radiations associated with the motion of free electrons (**continuous X-rays**) or bound electrons (**characteristic X-rays**). In the following discussion the term "gamma ray" will be used, but it is to be understood that what is said about interaction with matter applies identically to X-rays.

Nature of the Attenuation Laws

Gamma-ray photons all travel with the velocity of light. They have intrinsic energy, usually measured in Mev. The probabilities that the photon will interact in any of several ways with the material through which it passes are unique functions of the photon energy. If the photon does not interact, it passes through the material with no effect. Because of the statistical nature of interaction probabilities, there will always be a finite probability of penetration of any shield; so that there is no such phenomenon as complete attenuation in the sense that it is possible to achieve, for example, watertightness.

Attenuation is accomplished by interactions which either absorb the photon outright, reduce its energy and change its direction, or in some cases merely change its direction. The **probability for photon interaction** in a given material is expressed as probability per unit path length. In other words, given a "current density," or flux of photons of strength I per square centimeter per second, the fractional decrease per unit path length would be a constant, called the **attenuation coefficient** μ , cm^{-1} :

$$\frac{dI/dx}{I} = \mu \quad (7-3)$$

or

$$I = I_0 e^{-\mu x} \quad (7-3a)$$

where I_0 is the flux at $x = 0$. Here I refers to "uncollided" photons, and μ is the

total attenuation coefficient; μ is often referred to also as the **narrow-beam attenuation coefficient**, since in an attenuation measurement with a narrow beam all photons which have interacted, whether by absorption or deflection, are removed from the beam. Sometimes it is convenient to express the path in terms of the mass of material traversed, which is just the product of the path length and the density ρ . In this case the **mass-attenuation coefficient** μ/ρ , cm^2/g , is used. The advantage of using μ/ρ to characterize attenuation of an element is clear when it is realized that the probability of interaction is proportional to the amount of material traversed; so that the quantity μ/ρ is characteristic of the composition of the material but is independent of the density and physical (or chemical) form.

The probability of interaction with one element (say, the i th element) of a material (compound, mix, etc.) is independent of the presence of other elements; so that the total probability of interaction is the simple sum of probabilities for all elements of the material. Thus,

$$\mu = \sum_i (\mu/\rho)_i \rho_i \quad (7-4)$$

where ρ_i = density, g/cm^3 , of the i th constituent of the material as it appears in the material (weight fraction times material density)

$(\mu/\rho)_i$ = mass-attenuation coefficient, cm^2/g , for the i th constituent.

In case two different materials, 1 and 2, are to be traversed successively, the attenuation is the product of the individual attenuations, i.e.,

$$I = I_0 e^{-\mu_1 x_1} e^{-\mu_2 x_2} \quad (7-5)$$

$$= I_0 e^{-(\mu_1 x_1 + \mu_2 x_2)} \quad (7-5a)$$

Interaction Processes

As was mentioned above, the interaction processes can have any of three effects on the photons:

1. Outright absorption: The photon disappears, giving its energy to an electron or to an electron-positron pair. Other photons may subsequently appear, but with much less energy and more or less randomly oriented.

2. Scattering with reduction in energy: the lost energy is given to an electron.

3. Scattering without appreciable energy loss.

The individual interaction processes which cause these effects are described below.

Photoelectric Effect. A photon can interact with a bound electron in an atom, disappearing itself, and causing the electron to leave the atom. The electron leaves with energy nearly equal to the difference between the original photon energy and the binding energy with which it was held in the atom. The **cross section**, or probability per atom, for this phenomenon varies from element to element approximately as Z^5 (Z is the atomic number). It decreases very rapidly with increasing photon energy for a given element (about as E^{-3} for $E < 0.5$ Mev and E^{-1} for $E > 0.5$). The most likely electron to be dislodged is that which is most tightly bound in the atom, provided the photon had sufficient energy to "unbind" it. Since the atom must subsequently readjust itself, this means that there may be subsequent electromagnetic radiation from the atom as an outer electron falls into the inner orbit. This "fluorescent radiation" may escape from the atom, or it may cause ejection of a second electron, usually a loosely bound one. The fluorescent radiation is seldom

important biologically by comparison with the original gamma radiation. Consequently, the photoelectric process is generally thought of as a genuine absorption process, even though there may be a low-energy photon leaving the interacting atom.

Compton Process. A photon can interact with a loosely bound electron, sharing its energy and momentum; so that the electron is accelerated and the photon is deflected with lower energy. The probability per unit path length for this process is proportional to the density of electrons; so that the cross section per atom is proportional to Z . The most probable Compton interactions are those in which the photon is deflected and degraded in energy least. This process is one of scattering and not so effective in attenuation as the photoelectric effect. The cross section decreases monotonically with increasing photon energy. The energy degradation and the angle of scattering are connected by an expression which ensures simultaneous conservation of energy and momentum in the interaction process:

$$E' = \frac{E}{1 + 1.96E(1 - \cos \theta)}$$

where E' , E = photon energy after and before interaction, respectively, Mev
 θ = angle of deflection of photon

Pair Production. A photon with energy greater than 1.02 Mev can interact with the coulomb field of an atom (usually the strong field near the nucleus, although it can interact with that of an electron), producing an electron-positron pair. The photon disappears. The positron shortly thereafter is annihilated, with the emission of a total of about 1.02 Mev, usually equally divided between two photons. This process is more or less tantamount to absorption, since the high-energy photon disappears and is replaced by lower-energy photons which are more easily attenuated by the other processes. The cross section for this process varies from element to element approximately as Z^2 . It is a monotonically increasing function of photon energy up to about 50 Mev for high- Z elements, and to an even higher energy for low- Z elements.

Rayleigh Scattering. Low-energy photons are scattered elastically, through small angles, with negligible degradation in energy. This process is unimportant in shielding and biology.

Relative Importance of the Interaction Processes

The "total attenuation coefficient," or just **attenuation coefficient**, is the sum of those for all three processes, i.e.,

$$\mu_{\text{total}} = \mu_{pe} + \mu_c + \mu_{pp}$$

where μ_{pe} = attenuation coefficient for photoelectric effect

μ_c = attenuation coefficient for Compton effect

μ_{pp} = attenuation coefficient for pair production

For low-energy photons ($E < \sim 1$ Mev) and high- Z elements, the photoelectric-effect cross section is very high, and this process is strongly dominant. For lower- Z elements it becomes less important, being essentially negligible in hydrogen except for extremely low-energy photons ($E < 0.01$ Mev).

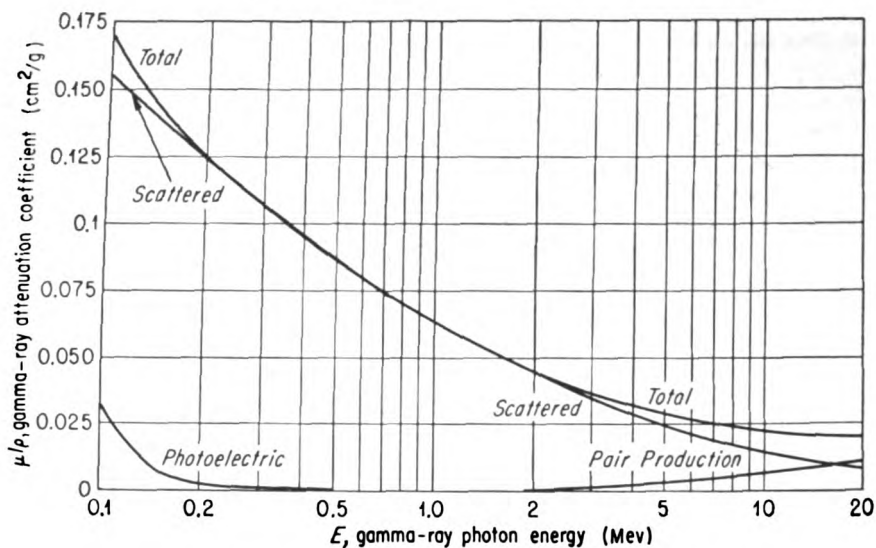


FIG. 7-1. Gamma-ray attenuation coefficients for ordinary concrete.

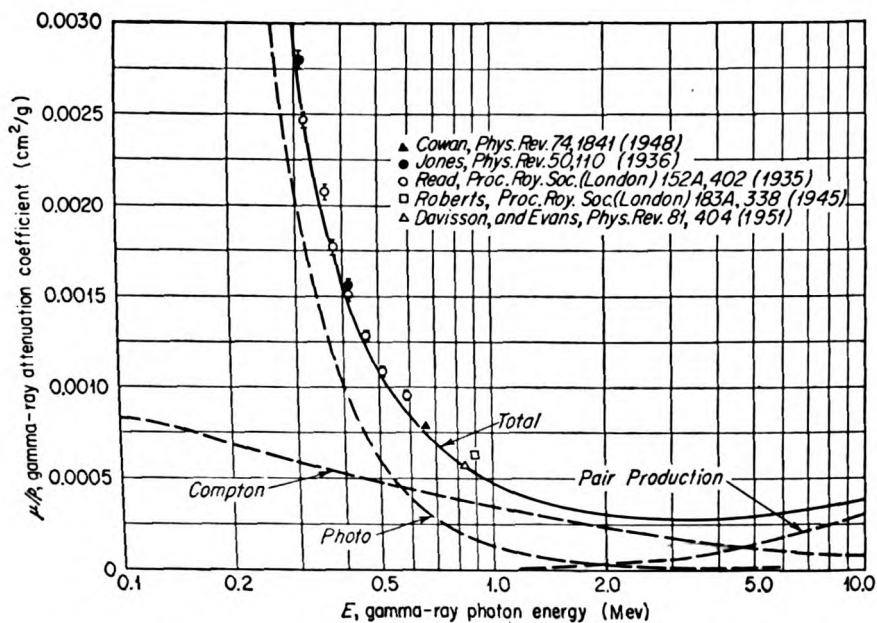


FIG. 7-2. Gamma-ray attenuation coefficients for lead. (From Davisson and Evans, *Revs. Mod. Phys.*, vol. 24, p. 79, 1952.)

For intermediate photon energies ($1 < E < 5$ Mev) with essentially all elements, the Compton effect is very important, even taking into consideration that it is merely a scattering process and not an absorption process. For lower- Z elements, the upper energy at which the foregoing is valid increases strongly (10 Mev for aluminum).

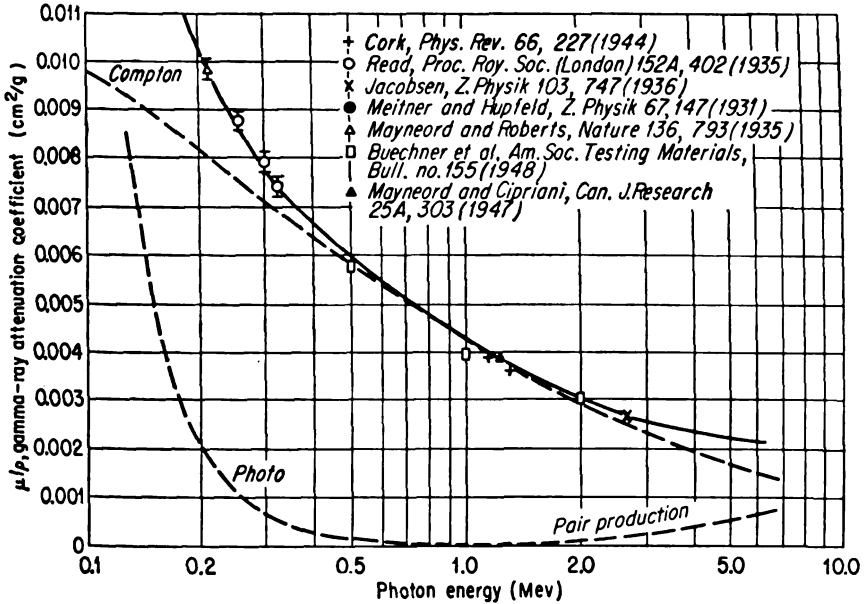


Fig. 7-3. Gamma-ray attenuation coefficients for iron. (From Davisson and Evans, *Revs. Mod. Phys.*, vol. 24, p. 79, 1952.)

For high-energy photons ($E > 5$ Mev) in high- Z elements the pair-production effect dominates. In lower- Z elements this process becomes dominant only at higher energies.

The relative importance of these three major processes for typical shield materials is seen in Figs. 7-1 to 7-3.

Energy Absorption from Gamma Rays

The rate of removal of energy from the *uncollided* beam (Mev per cm^3 sec) is the product of I , E , and μ_{total} . In order to estimate the rate of **energy deposition** from a flux of **gamma rays**, it is necessary to subtract off from this product the energy carried away by scattered photons or by other secondary photons. The principal contribution to this diminution effect on energy deposition is the Compton-scattered photons. If the other secondary escape effects, such as fluorescence, are neglected, the **energy absorption coefficient**, or **broad-beam attenuation coefficient**, would be

$$\mu_a = \mu_{pe} + \left(\frac{E - E'}{E} \right) \mu_c + \mu_{pp}$$

where $\left(\frac{E - E'}{E}\right)$ = average fractional energy loss in a Compton scattering of a photon of energy E

In this case the energy deposition rate, Mev/cm³ sec, would be given by the product of the photon flux I , the photon energy E , and μ_a . The quantity μ_a/ρ is usually tabulated, and use of this quantity instead of μ_a in the calculation would give energy deposition rate per unit mass of material, a useful quantity in the study of biological effects.

Build-up Factors

While it is convenient for purposes of explanation to discuss "uncollided flux" and to refer to flux as if all photons had the same energy, this is not very useful in a practical sense, because after attenuation of a gamma-ray flux the Compton process will have introduced other photons of lower energy which must also be considered. Thus every physically observable effect—energy deposition, number and energy flux, biological dose—will be enhanced by the action of these scattered photons. If the effect attributable to uncollided photons, a function in general easily calculated, is considered as a base, then the total effect, due to all photons, will be greater than this base value by a quantity called the **build-up factor**. This factor will vary with the source photon energy, the kind of attenuating material and its thickness, and the quantity being observed (e.g., dose, flux, etc.). Build-up factors for biological dose and for energy deposition (heating) have been tabulated for most elements and for water (Goldstein and Wilkins, Calculations of the Penetration of Gamma Rays, *AEC Rept.* NYO-3075, 1954).

NEUTRONS

Bibliography: Evans, "The Atomic Nucleus," chap. 14, McGraw-Hill, 1955.

Classification of Neutrons

Neutrons are classed according to their kinetic energy because in general this parameter determines the probability for interaction with any given matter. While an exact specification of energy is really necessary for accurate assay of interaction probabilities, it is convenient to specify ranges of energy within which the interactions vary in a regular way.

Thermal Neutrons. Thermal neutrons are those which are in thermal equilibrium with surrounding matter, so that on the average there is no net exchange of kinetic energy between neutrons and the thermally agitated atoms of the matter. Unless otherwise stated, the matter is assumed to be at room temperature. In this case, the neutrons will have a Maxwellian distribution of velocities, and in this velocity distribution the most probable velocity is 2.2×10^5 cm/sec, corresponding to a kinetic energy of 0.025 ev.

Epithermal Neutrons. As the name implies, this range of energies includes those just above the thermal range (0.2 to about 10 ev). They are characterized by having enough energy so that they are not strongly absorbed in cadmium (the term

epicadmium is often used), but not sufficient energy to be important, say, in knocking atoms out of lattices.

Intermediate Neutrons. This energy range is variously described. Many authors define it as between 0.5 and 10,000 ev, this being an interesting range in nuclear physics which can be studied in detail in the laboratory. It is characterized by **resonance peaks** in interaction cross sections, these being narrow energy bands for which the interaction probabilities are abnormally high. For this reason the neutrons in this energy range are often called **resonance neutrons**. For different reasons, in shielding work the intermediate-energy region extends up to about 0.3 Mev, or to the inelastic-scattering (see below) threshold.

Fast Neutrons. Neutrons of energy greater than intermediate energy, but less than 10 Mev, are called fast neutrons. This includes almost all virgin neutrons from the fission process.

Relativistic Neutrons. Above about 10 Mev the kinetic energy becomes a significant fraction of the total (kinetic plus rest) energy of the neutrons, so that relativistic corrections should be applied in treatments of neutron interaction.

Interaction Processes

Interaction processes (Hughes and Harvey, Neutron Cross Sections, *Brookhaven Natl. Lab. Rept.* 325, 1955) are specified in terms of a neat code, which gives the entering and departing nuclear particles, written consecutively in parentheses, separated by a comma. Thus the interaction of a neutron with a nucleus, following which a proton is released, is written (n,p) . If the reacting and product nuclei are to be written also, the code gives $N-14(n,p)C-14$.

Elastic Scattering (n,n) . The most common interaction of neutrons with matter is elastic scattering. Since the interaction is elastic, the sum of the kinetic energies of neutron and struck atom has the same total before as after the event. If the struck atom is considered initially stationary, the ratio of neutron energy after collision to that before is

$$\frac{E_2}{E_1} = \frac{A^2 + 2A \cos \theta + 1}{(A + 1)^2} \quad (7-6)$$

where A = atomic weight of struck atom

θ = angle of deflection of neutron in the center-of-mass system

(For very light nuclei, the angle of deflection in the laboratory system will be significantly smaller, being half as large for hydrogen.)

The distribution in angle θ of the scattered neutrons is essentially isotropic for neutrons below $\frac{1}{4}$ Mev. Thus the probability of scattering into an element of solid angle is proportional only to the size of the element and independent of the angle θ . For neutrons above this energy, isotropic scattering no longer holds, the forward directions becoming significantly more probable (Hughes and Carter, Neutron Cross Sections—Angular Distributions, *Brookhaven Natl. Lab. Rept.* 400, 1956).

Inelastic Scattering $(n,n'\gamma)$. In inelastic scattering the neutron interacts with the nucleus, raises it to an excited state, and then departs with correspondingly lower energy. The excited nucleus at once decays to the ground state with the emission of one or more gamma-ray photons. Since the neutron must have sufficient

energy so that it can raise the struck nucleus to at least the lowest permissible excited state (this will of course differ from nucleus to nucleus), there is a threshold neutron energy below which this cannot take place. For the larger nuclei it is of the order of 0.1 Mev, and for some light nuclei very much higher. In general, the neutron energy loss on inelastic scattering is very great, especially if its initial energy is well above the threshold.

Neutron Capture (n,γ). In neutron capture, the neutron joins the struck nucleus essentially permanently. The new nucleus so formed is then excited by an amount equal to the neutron binding energy (plus whatever excess might accrue from its kinetic energy, which is usually small). The binding energy varies from nucleus to nucleus, averaging about 8 Mev for most elements. Notable exceptions are hydrogen (2.23 Mev) and nitrogen (10.8 Mev). Neutron capture is significant only for low-energy neutrons (generally, thermal and epithermal energies). The cross section for capture varies with neutron energy in two ways. One component is proportional to the reciprocal of the neutron velocity ($1/v$, or $1/\sqrt{E}$). The other is evidenced as sharp resonances at discrete energies. A strong capture resonance at thermal energy is exhibited by cadmium.

Charged-particle Reactions (n,p); (n,α); ($n,\alpha\gamma$). In these reactions a neutron enters the nucleus and a charged particle (e.g., proton, alpha particle, deuteron, or triton) emerges. In some cases a gamma ray may also come off ($n,\alpha\gamma$). These processes are generally important only for fast neutrons, but there are several important exceptions, all exoergic, which are significant for thermal neutrons. These are given in Table 7-2.

Table 7-2. Important Thermal-neutron Charged-particle Reactions

Struck nucleus	Reaction	Thermal-neutron cross section, barns		Energy release Q , Mev
		Natural element	Isotope	
Li ⁶	Li ⁶ (n,α)H ³	70	930	4.78
B ¹⁰	B ¹⁰ (n,α)Li ⁷	50	266	2.792
	B ¹⁰ ($n,\alpha\gamma$)Li ⁷	700	3,720	2.792 *
N ¹⁴	N ¹⁴ (n,p)C ¹⁴	1.68	1.68	0.624

* The gamma ray carries off 0.48 Mev.

Other Reactions. If the incident neutron has sufficient energy, it can cause the emission of two or more neutrons (or other particles). The neutron can also induce fission in certain heavy nuclei. These two reactions are denoted as ($n,2n$) and (n,f), respectively.

General References

- Segrè (ed.), "Experimental Nuclear Physics," Wiley, 1953.
 Evans, "The Atomic Nucleus," McGraw-Hill, 1955.
 Feather, *Proc. Cambridge Phil. Soc.*, vol. 34, p. 599, 1938.

- Marshall and Ward, *Can. J. Research*, vol. A15, p. 39, 1937.
- Glendinin, *Nucleonics*, vol. 2, p. 12, 1948.
- Flammersfeld, *Nature*, vol. 33, p. 280, 1946.
- Bleuler and Zunti, *Helv. Phys. Acta*, vol. 19, p. 137, 1946.
- Bethe, *Ann. Physik*, vol. 5, p. 325, 1930.
- Bethe, "Handbuch der Physik," vol. 24, p. 273, Springer, Berlin, 1953.
- Bleuler and Zunti, *Helv. Phys. Acta*, vol. 19, p. 383, 1946.
- White, X-ray Attenuation Coefficients, *Natl. Bur. Standards (U.S.) Rept.* 1003, 1942.
- Davissan and Evans, *Revs. Modern Phys.*, vol. 24, p. 79, 1952.
- Fano, U., *Nucleonics*, vol. 11, no. 8, p. 8, 1953.
- U.S. Atomic Energy Commission, "Reactor Handbook," vol. 1, Sec. 2, *AECD Rept.* 3645, 1955.
- Goldstein and Wilkins, Calculations of the Penetration of Gamma Rays, *AEC Rept.* NYO-3075, 1954.
- Hughes and Harvey, Neutron Cross Sections, *Brookhaven Natl. Lab. Rept.* 325, 1955.
- Hughes and Carter, Neutron Cross Sections—Angular Distributions, *Brookhaven Natl. Lab. Rept.* 400, 1956.

Section 8

RADIATION ATTENUATION DATA

By

HAROLD O. WYCKOFF, Ph.D., *National Bureau of Standards, Washington, D.C.*

X- and Gamma-ray Attenuation Data.....	8-2
Elements.....	8-4
Water.....	8-28
Sodium Iodide.....	8-29
Calcium Phosphate.....	8-30
Air.....	8-31
Concrete.....	8-31
Build-up Factors.....	8-33
Water (Point Source).....	8-33
Aluminum (Point Source).....	8-33
Iron (Point Source).....	8-33
Tin (Point Source).....	8-34
Tungsten (Point Source).....	8-34
Lead (Point Source).....	8-34
Water (Plane Source).....	8-35
Iron (Plane Source).....	8-35
Tin (Plane Source).....	8-35
Lead (Plane Source).....	8-36
Uranium (Plane Source).....	8-36
Attenuation Computations.....	8-37
Attenuation Curves.....	8-39
Shielding Materials.....	8-41
Shielding Computations.....	8-43

RADIATION ATTENUATION DATA

Harold O. Wyckoff

X- AND GAMMA-RAY ATTENUATION DATA

When a barrier is placed in a beam of radiation, the amount transmitted through the barrier depends not only on the thickness of the barrier but also on the amount of scattered radiation that reaches the measuring instrument. Figure 8-1 illustrates this point. *Two limiting cases* are easily defined. The **narrow-beam** or good-

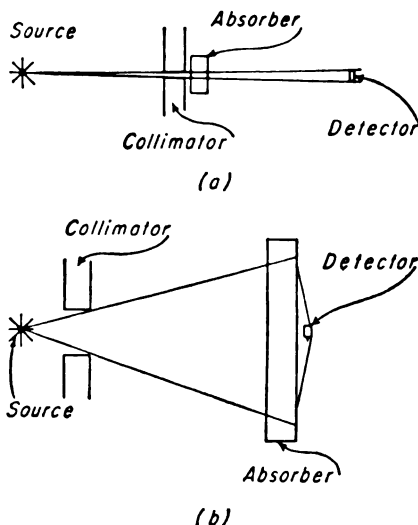


FIG. 8-1. Comparison of possible experimental arrangements for (a) narrow- and (b) broad-beam geometry.

geometry condition is obtained when none of the scattered radiation is measured. This condition is obtained experimentally by very small irradiated areas of the absorbing material. Usually an extrapolation is required for most precise determination. The **broad-beam** or poor-geometry condition is obtained when the instrument measures a maximum amount of scattered radiation. The latter condition is obtained experimentally either by broad irradiated areas of a flat barrier and/or close proximity of the source and/or detector to the barrier, or for spherical barriers

around the source or detector. Some differences may arise from these various geometrical conditions.

Figure 8-2 shows the attenuation curves for the two limiting cases with Cobalt-60 gamma rays. The attenuation curve for a monoenergetic photon source and narrow-beam conditions may be expressed mathematically as

$$\frac{I}{I_0} = e^{-\mu x} \quad (8-1)$$

where μ is the total absorption coefficient of the incident beam and I and I_0 are measures of the radiation intensity with barrier thickness X and zero, respectively.

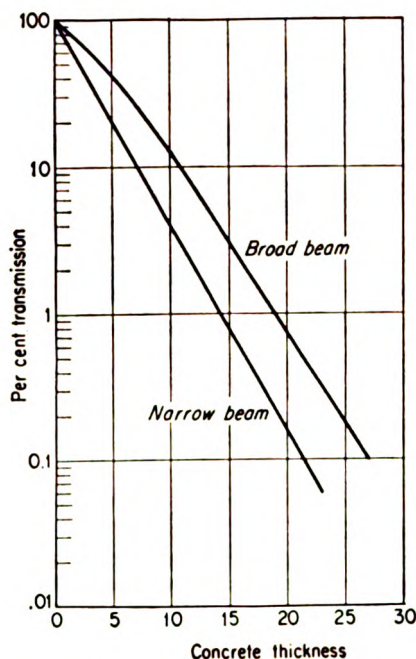


FIG. 8-2. Broad- and narrow-beam attenuation curves in concrete for gamma rays from Cobalt-60.

The total absorption coefficient is made up of the fraction of intensity given to the Compton electron σ_a , scattered photon σ_s , to the photoelectric process τ , and to the pair-production process κ . See Tables 8-1 to 8-29 for recently tabulated values. (Grodstein, Gladys White, *Nat. Bur. Standards Circ. 583*, GPO, Washington, April 30, 1957.) For broad-beam conditions, the expression becomes

$$\frac{I}{I_0} = Be^{-\mu x} \quad (8-2)$$

where B is the **build-up factor**. The ratio of the transmitted intensity for broad-beam to that for narrow-beam conditions is called the build-up factor for that

Table 8-1. Hydrogen

Photon energy, Mev	Scattering * without coherent, barns/atom	Photo-electric ls electron, barns/atom	Pair production		Total † without coherent, cm ² /g
			Nucleus, barns/atom	Electron, barns/atom	
0.01	0.640	0.0046			0.385
0.015	0.629	0.0011			0.377
0.02	0.618				0.369
0.03	0.597				0.357
0.04	0.578				0.345
0.05	0.561				0.335
0.06	0.546				0.326
0.08	0.517				0.309
0.10	0.493				0.295
0.15	0.444				0.265
0.20	0.407				0.243
0.30	0.354				0.212
0.40	0.317				0.189
0.50	0.289				0.173
0.60	0.268				0.160
0.80	0.235				0.140
1.0	0.211				0.126
1.5	0.1716		0.000044		0.103
2.0	0.1464		0.00018		0.0876
3.0	0.1151		0.00051	0.00001	0.0691
4.0	0.0960		0.00082	0.00005	0.0579
5.0	0.0828		0.0011	0.0001	0.0502
6.0	0.0732		0.0013	0.0002	0.0446
8.0	0.0599		0.0018	0.0004	0.0371
10	0.0510		0.0021	0.0006	0.0321
15	0.0377		0.0028	0.0011	0.0249
20	0.0302		0.0033	0.0015	0.0209
30	0.0220		0.0040	0.0021	0.0168
40	0.01746		0.0045	0.0026	0.0147
50	0.01456		0.0048	0.0029	0.0133
60	0.01254		0.0051	0.0033	0.0125
80	0.00988		0.0056	0.0038	0.0115
100	0.00820		0.0059	0.0042	0.0109

* Total scattering for hydrogen is given by the Klein-Nishina formula for free electrons.

† Barns/atom $\times 0.5997 = \text{cm}^2/\text{g}$.

Table 8-2. Beryllium

Photon energy, Mev	Scattering *		Photo-electric <i>K</i> and <i>L</i> shells, barns/atom	Pair production		Total †	
	With coherent, barns/atom	Without coherent, barns/atom		Nucleus, barns/atom	Electron, barns/atom	With coherent, cm ² /g	Without coherent, cm ² /g
0.01	3.54	2.56	5.42			0.599	0.533
0.015	3.01	2.52	1.39			0.294	0.261
0.02	2.77	2.47	0.52			0.220	0.200
0.03	2.53	2.39	0.13			0.178	0.168
0.04	2.38	2.31	0.052			0.163	0.158
0.05	2.29	2.24	0.021			0.154	0.151
0.06	2.21	2.18	0.010			0.148	0.146
0.08	2.10	2.07				0.140	0.138
0.10	1.99	1.972				0.133	0.132
0.15	1.78	1.774				0.119	0.119
0.20	1.63	1.626				0.109	0.109
0.30		1.414					0.0945
0.40		1.267					0.0847
0.50		1.157					0.0773
0.60		1.070					0.0715
0.80		0.940					0.0628
1.0		0.845					0.0565
1.5		0.686		0.00071			0.0459
2.0		0.586		0.0028			0.0394
3.0		0.460		0.0081	0.00005		0.0313
4.0		0.384		0.013	0.0002		0.0266
5.0		0.331		0.018	0.0004		0.0234
6.0		0.293		0.022	0.0008		0.0211
8.0		0.240		0.028	0.002		0.0180
10		0.204		0.034	0.003		0.0161
15		0.1509		0.044	0.004		0.0133
20		0.1210		0.052	0.006		0.0120 ‡
30		0.0880		0.063	0.008		0.0106
40		0.0698		0.070	0.010		0.0100
50		0.0582		0.076	0.012		0.00977
60		0.0502		0.081	0.013		0.00964
80		0.0395		0.087	0.015		0.00946
100		0.0328		0.093	0.017		0.00955

* Data in the first column are given by the sum of coherent scattering and of incoherent scattering from the Klein-Nishina formula corrected for binding effects. In the second column incoherent scattering is given by the Klein-Nishina formula for free electrons.

† Barns/atom $\times 0.06684 = \text{cm}^2/\text{g}$.

‡ Energy region in which dipole absorption attains a maximum cross section.

Table 8-3. Carbon

Photon energy, Mev	Scattering *		Photo-electric K and L shells, barns/atom	Pair production		Total †	
	With coherent, barns/atom	Without coherent, barns/atom		Nucleus, barns/atom	Electron, barns/atom	With coherent, cm ² /g	Without coherent, cm ² /g
0.01	6.88	3.84	38.6			2.28	2.13
0.015	5.30	3.77	10.2			0.777	0.701
0.02	4.64	3.71	3.91			0.429	0.382
0.03	4.04	3.58	0.99			0.252	0.229
0.04	3.71	3.47	0.38			0.205	0.193
0.05	3.50	3.37	0.18			0.185	0.178
0.06	3.37	3.28	0.096			0.174	0.169
0.08	3.18	3.10	0.037			0.161	0.157
0.10	3.02	2.96	0.017			0.152	0.149
0.15	2.69	2.66	0.0040			0.135	0.134
0.20	2.46	2.44				0.123	0.122
0.30	2.13	2.12				0.107	0.106
0.40		1.900					0.0953
0.50		1.735					0.0870
0.60		1.605					0.0805
0.80		1.410					0.0707
1.0		1.267					0.0636
1.5		1.030		0.0016			0.0518
2.0		0.878		0.0063			0.0444
3.0		0.691		0.018	0.00007		0.0366
4.0		0.576		0.030	0.0003		0.0304
5.0		0.497		0.040	0.0007		0.0270
6.0		0.439		0.048	0.001		0.0245
8.0		0.359		0.063	0.002		0.0213
10		0.306		0.076	0.004		0.0194
15		0.226		0.099	0.006		0.0166
20		0.1814		0.116	0.009		0.0154 †
30		0.1319		0.140	0.012		0.0142
40		0.1048		0.157	0.015		0.0139
50		0.0874		0.170	0.018		0.0138
60		0.0752		0.180	0.020		0.0138
80		0.0593		0.195	0.023		0.0139
100		0.0492		0.207	0.025		0.0141

* Data in the first column are given by the sum of coherent scattering and of incoherent scattering from the Klein-Nishina formula corrected for binding effects. In the second column incoherent scattering is given by the Klein-Nishina formula for free electrons.

† Barns/atom $\times 0.05016 = \text{cm}^2/\text{g}$.

‡ Energy region in which dipole absorption attains a maximum cross section.

Table 8-4. Nitrogen

Photon energy, Mev	Scattering *		Photo-electric <i>K</i> and <i>L</i> shells, barns/atom	Pair production		Total †	
	With coherent, barns/atom	Without coherent, barns/atom		Nucleus, barns/atom	Electron, barns/atom	With coherent, cm ² /g	Without coherent, cm ² /g
0.01	8.96	4.48	79.4			3.80	3.61
0.015	6.72	4.40	21.2			1.20	1.10
0.02	5.73	4.33	8.21			0.600	0.539
0.03	4.84	4.18	2.15			0.301	0.272
0.04	4.45	4.05	0.81			0.226	0.209
0.05	4.14	3.93	0.38			0.194	0.185
0.06	3.98	3.82	0.21			0.180	0.173
0.08	3.73	3.62	0.082			0.164	0.159
0.10	3.54	3.45	0.041			0.154	0.150
0.15	3.15	3.11	0.010			0.136	0.134
0.20	2.87	2.85				0.123	0.123
0.30	2.48	2.47				0.107	0.106
0.40		2.22					0.0955
0.50		2.02					0.0869
0.60		1.872					0.0805
0.80		1.645					0.0707
1.0		1.478					0.0636
1.5		1.201		0.0022			0.0517
2.0		1.025		0.0086			0.0445
3.0		0.806		0.025	0.00009		0.0357
4.0		0.672		0.040	0.0003		0.0306
5.0		0.580		0.054	0.0008		0.0273
6.0		0.512		0.066	0.001		0.0249
8.0		0.419		0.086	0.003		0.0218
10		0.357		0.103	0.004		0.0200
15		0.264		0.134	0.008		0.0175
20		0.212		0.158	0.010		0.0163
30		0.1539		0.190	0.015		0.0154 ‡
40		0.1222		0.213	0.018		0.0152
50		0.1019		0.231	0.020		0.0152
60		0.0878		0.244	0.023		0.0153
80		0.0692		0.264	0.026		0.0154
100		0.0574		0.280	0.029		0.0158

* Data in the first column are given by the sum of coherent scattering and of incoherent scattering from the Klein-Nishina formula corrected for binding effects. In the second column incoherent scattering is given by the Klein-Nishina formula for free electrons.

† Barns/atom $\times 0.04301 = \text{cm}^2/\text{g}$.

‡ Energy region in which dipole absorption attains a maximum cross section.

where $\left(\frac{E - E'}{E}\right)$ = average fractional energy loss in a Compton scattering of a photon of energy E

In this case the energy deposition rate, Mev/cm³ sec, would be given by the product of the photon flux I , the photon energy E , and μ_a . The quantity μ_a/ρ is usually tabulated, and use of this quantity instead of μ_a in the calculation would give energy deposition rate per unit mass of material, a useful quantity in the study of biological effects.

Build-up Factors

While it is convenient for purposes of explanation to discuss "uncollided flux" and to refer to flux as if all photons had the same energy, this is not very useful in a practical sense, because after attenuation of a gamma-ray flux the Compton process will have introduced other photons of lower energy which must also be considered. Thus every physically observable effect—energy deposition, number and energy flux, biological dose—will be enhanced by the action of these scattered photons. If the effect attributable to uncollided photons, a function in general easily calculated, is considered as a base, then the total effect, due to all photons, will be greater than this base value by a quantity called the **build-up factor**. This factor will vary with the source photon energy, the kind of attenuating material and its thickness, and the quantity being observed (e.g., dose, flux, etc.). Build-up factors for biological dose and for energy deposition (heating) have been tabulated for most elements and for water (Goldstein and Wilkins, *Calculations of the Penetration of Gamma Rays*, AEC Rept. NYO-3075, 1954).

NEUTRONS

Bibliography: Evans, "The Atomic Nucleus," chap. 14, McGraw-Hill, 1955.

Classification of Neutrons

Neutrons are classed according to their kinetic energy because in general this parameter determines the probability for interaction with any given matter. While an exact specification of energy is really necessary for accurate assay of interaction probabilities, it is convenient to specify ranges of energy within which the interactions vary in a regular way.

Thermal Neutrons. Thermal neutrons are those which are in thermal equilibrium with surrounding matter, so that on the average there is no net exchange of kinetic energy between neutrons and the thermally agitated atoms of the matter. Unless otherwise stated, the matter is assumed to be at room temperature. In this case, the neutrons will have a Maxwellian distribution of velocities, and in this velocity distribution the most probable velocity is 2.2×10^5 cm/sec, corresponding to a kinetic energy of 0.025 ev.

Epithermal Neutrons. As the name implies, this range of energies includes those just above the thermal range (0.2 to about 10 ev). They are characterized by having enough energy so that they are not strongly absorbed in cadmium (the term

epicadmium is often used), but not sufficient energy to be important, say, in knocking atoms out of lattices.

Intermediate Neutrons. This energy range is variously described. Many authors define it as between 0.5 and 10,000 ev, this being an interesting range in nuclear physics which can be studied in detail in the laboratory. It is characterized by **resonance peaks** in interaction cross sections, these being narrow energy bands for which the interaction probabilities are abnormally high. For this reason the neutrons in this energy range are often called **resonance neutrons**. For different reasons, in shielding work the intermediate-energy region extends up to about 0.3 Mev, or to the inelastic-scattering (see below) threshold.

Fast Neutrons. Neutrons of energy greater than intermediate energy, but less than 10 Mev, are called fast neutrons. This includes almost all virgin neutrons from the fission process.

Relativistic Neutrons. Above about 10 Mev the kinetic energy becomes a significant fraction of the total (kinetic plus rest) energy of the neutrons, so that relativistic corrections should be applied in treatments of neutron interaction.

Interaction Processes

Interaction processes (Hughes and Harvey, Neutron Cross Sections, *Brookhaven Natl. Lab. Rept.* 325, 1955) are specified in terms of a neat code, which gives the entering and departing nuclear particles, written consecutively in parentheses, separated by a comma. Thus the interaction of a neutron with a nucleus, following which a proton is released, is written (n,p) . If the reacting and product nuclei are to be written also, the code gives $N-14(n,p)C-14$.

Elastic Scattering (n,n) . The most common interaction of neutrons with matter is elastic scattering. Since the interaction is elastic, the sum of the kinetic energies of neutron and struck atom has the same total before as after the event. If the struck atom is considered initially stationary, the ratio of neutron energy after collision to that before is

$$\frac{E_2}{E_1} = \frac{A^2 + 2A \cos \theta + 1}{(A + 1)^2} \quad (7-6)$$

where A = atomic weight of struck atom

θ = angle of deflection of neutron in the center-of-mass system

(For very light nuclei, the angle of deflection in the laboratory system will be significantly smaller, being half as large for hydrogen.)

The distribution in angle θ of the scattered neutrons is essentially isotropic for neutrons below $\frac{1}{4}$ Mev. Thus the probability of scattering into an element of solid angle is proportional only to the size of the element and independent of the angle θ . For neutrons above this energy, isotropic scattering no longer holds, the forward directions becoming significantly more probable (Hughes and Carter, Neutron Cross Sections—Angular Distributions, *Brookhaven Natl. Lab. Rept.* 400, 1956).

Inelastic Scattering $(n,n'\gamma)$. In inelastic scattering the neutron interacts with the nucleus, raises it to an excited state, and then departs with correspondingly lower energy. The excited nucleus at once decays to the ground state with the emission of one or more gamma-ray photons. Since the neutron must have sufficient

energy so that it can raise the struck nucleus to at least the lowest permissible excited state (this will of course differ from nucleus to nucleus), there is a threshold neutron energy below which this cannot take place. For the larger nuclei it is of the order of 0.1 Mev, and for some light nuclei very much higher. In general, the neutron energy loss on inelastic scattering is very great, especially if its initial energy is well above the threshold.

Neutron Capture (n,γ). In neutron capture, the neutron joins the struck nucleus essentially permanently. The new nucleus so formed is then excited by an amount equal to the neutron binding energy (plus whatever excess might accrue from its kinetic energy, which is usually small). The binding energy varies from nucleus to nucleus, averaging about 8 Mev for most elements. Notable exceptions are hydrogen (2.23 Mev) and nitrogen (10.8 Mev). Neutron capture is significant only for low-energy neutrons (generally, thermal and epithermal energies). The cross section for capture varies with neutron energy in two ways. One component is proportional to the reciprocal of the neutron velocity ($1/v$, or $1/\sqrt{E}$). The other is evidenced as sharp resonances at discrete energies. A strong capture resonance at thermal energy is exhibited by cadmium.

Charged-particle Reactions (n,p); (n,α); ($n,\alpha\gamma$). In these reactions a neutron enters the nucleus and a charged particle (e.g., proton, alpha particle, deuteron, or triton) emerges. In some cases a gamma ray may also come off ($n,\alpha\gamma$). These processes are generally important only for fast neutrons, but there are several important exceptions, all exoergic, which are significant for thermal neutrons. These are given in Table 7-2.

Table 7-2. Important Thermal-neutron Charged-particle Reactions

Struck nucleus	Reaction	Thermal-neutron cross section, barns		Energy release Q , Mev
		Natural element	Isotope	
Li ⁶	Li ⁶ (n,α)H ³	70	930	4.78
B ¹⁰	B ¹⁰ (n,α)Li ⁷	50	266	2.792
	B ¹⁰ ($n,\alpha\gamma$)Li ⁷	700	3,720	2.792 *
N ¹⁴	N ¹⁴ (n,p)C ¹⁴	1.68	1.68	0.624

* The gamma ray carries off 0.48 Mev.

Other Reactions. If the incident neutron has sufficient energy, it can cause the emission of two or more neutrons (or other particles). The neutron can also induce fission in certain heavy nuclei. These two reactions are denoted as ($n,2n$) and (n,f), respectively.

General References

- Segrè (ed.), "Experimental Nuclear Physics," Wiley, 1953.
 Evans, "The Atomic Nucleus," McGraw-Hill, 1955.
 Feather, *Proc. Cambridge Phil. Soc.*, vol. 34, p. 599, 1938.

- Marshall and Ward, *Can. J. Research*, vol. A15, p. 39, 1937.
- Glendinin, *Nucleonics*, vol. 2, p. 12, 1948.
- Flammersfeld, *Nature*, vol. 33, p. 280, 1946.
- Bleuler and Zunti, *Helv. Phys. Acta*, vol. 19, p. 137, 1946.
- Bethe, *Ann. Physik*, vol. 5, p. 325, 1930.
- Bethe, "Handbuch der Physik," vol. 24, p. 273, Springer, Berlin, 1953.
- Bleuler and Zunti, *Helv. Phys. Acta*, vol. 19, p. 383, 1946.
- White, X-ray Attenuation Coefficients, *Natl. Bur. Standards (U.S.) Rept.* 1003, 1942.
- Davissan and Evans, *Revs. Modern Phys.*, vol. 24, p. 79, 1952.
- Fano, U., *Nucleonics*, vol. 11, no. 8, p. 8, 1953.
- U.S. Atomic Energy Commission, "Reactor Handbook," vol. 1, Sec. 2, *AECD Rept.* 3645, 1955.
- Goldstein and Wilkins, Calculations of the Penetration of Gamma Rays, *AEC Rept.* NYO-3075, 1954.
- Hughes and Harvey, Neutron Cross Sections, *Brookhaven Natl. Lab. Rept.* 325, 1955.
- Hughes and Carter, Neutron Cross Sections—Angular Distributions, *Brookhaven Natl. Lab. Rept.* 400, 1956.

Section 8

RADIATION ATTENUATION DATA

By

HAROLD O. WYCKOFF, Ph.D., *National Bureau of Standards, Washington, D.C.*

X- and Gamma-ray Attenuation Data.....	8-2
Elements.....	8-4
Water.....	8-28
Sodium Iodide.....	8-29
Calcium Phosphate.....	8-30
Air.....	8-31
Concrete.....	8-31
Build-up Factors.....	8-33
Water (Point Source).....	8-33
Aluminum (Point Source).....	8-33
Iron (Point Source).....	8-33
Tin (Point Source).....	8-34
Tungsten (Point Source).....	8-34
Lead (Point Source).....	8-34
Water (Plane Source).....	8-35
Iron (Plane Source).....	8-35
Tin (Plane Source).....	8-35
Lead (Plane Source).....	8-36
Uranium (Plane Source).....	8-36
Attenuation Computations.....	8-37
Attenuation Curves.....	8-39
Shielding Materials.....	8-41
Shielding Computations.....	8-43

RADIATION ATTENUATION DATA

Harold O. Wyckoff

X- AND GAMMA-RAY ATTENUATION DATA

When a barrier is placed in a beam of radiation, the amount transmitted through the barrier depends not only on the thickness of the barrier but also on the amount of scattered radiation that reaches the measuring instrument. Figure 8-1 illustrates this point. *Two limiting cases* are easily defined. The **narrow-beam** or good-

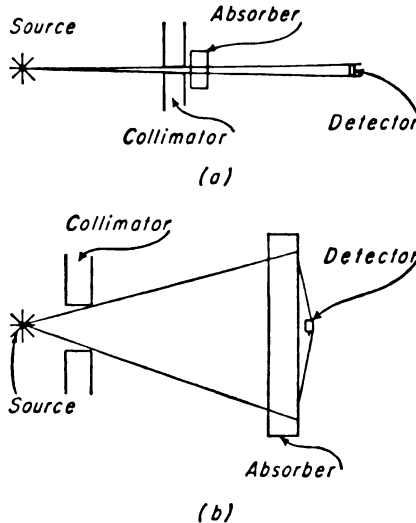


FIG. 8-1. Comparison of possible experimental arrangements for (a) narrow- and (b) broad-beam geometry.

geometry condition is obtained when none of the scattered radiation is measured. This condition is obtained experimentally by very small irradiated areas of the absorbing material. Usually an extrapolation is required for most precise determination. The **broad-beam** or poor-geometry condition is obtained when the instrument measures a maximum amount of scattered radiation. The latter condition is obtained experimentally either by broad irradiated areas of a flat barrier and/or close proximity of the source and/or detector to the barrier, or for spherical barriers

around the source or detector. Some differences may arise from these various geometrical conditions.

Figure 8-2 shows the attenuation curves for the two limiting cases with Cobalt-60 gamma rays. The attenuation curve for a monoenergetic photon source and narrow-beam conditions may be expressed mathematically as

$$\frac{I}{I_0} = e^{-\mu x} \quad (8-1)$$

where μ is the total absorption coefficient of the incident beam and I and I_0 are measures of the radiation intensity with barrier thickness X and zero, respectively.

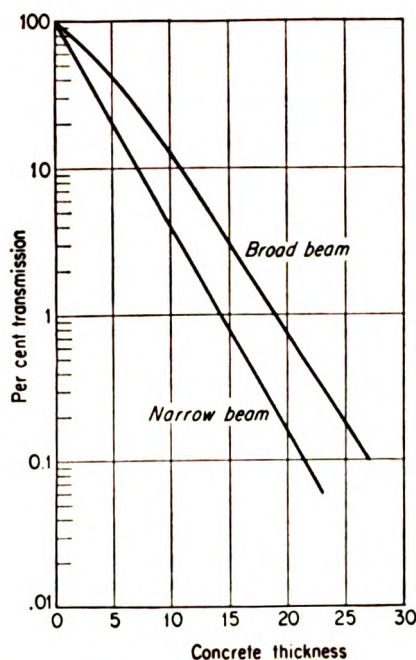


FIG. 8-2. Broad- and narrow-beam attenuation curves in concrete for gamma rays from Cobalt-60.

The total absorption coefficient is made up of the fraction of intensity given to the Compton electron σ_a , scattered photon σ_s , to the photoelectric process τ , and to the pair-production process κ . See Tables 8-1 to 8-29 for recently tabulated values. (Grodstein, Gladys White, *Nat. Bur. Standards Circ. 583*, GPO, Washington, April 30, 1957.) For broad-beam conditions, the expression becomes

$$\frac{I}{I_0} = Be^{-\mu x} \quad (8-2)$$

where B is the **build-up factor**. The ratio of the transmitted intensity for broad-beam to that for narrow-beam conditions is called the build-up factor for that

Table 8-1. Hydrogen

Photon energy, Mev	Scattering * without coherent, barns/atom	Photo-electric ls electron, barns/atom	Pair production		Total † without coherent, cm ² /g
			Nucleus, barns/atom	Electron, barns/atom	
0.01	0.640	0.0046			0.385
0.015	0.629	0.0011			0.377
0.02	0.618				0.369
0.03	0.597				0.357
0.04	0.578				0.345
0.05	0.561				0.335
0.06	0.546				0.326
0.08	0.517				0.309
0.10	0.493				0.295
0.15	0.444				0.265
0.20	0.407				0.243
0.30	0.354				0.212
0.40	0.317				0.189
0.50	0.289				0.173
0.60	0.268				0.160
0.80	0.235				0.140
1.0	0.211				0.126
1.5	0.1716		0.000044		0.103
2.0	0.1464		0.00018		0.0876
3.0	0.1151		0.00051	0.00001	0.0691
4.0	0.0960		0.00082	0.00005	0.0579
5.0	0.0828		0.0011	0.0001	0.0502
6.0	0.0732		0.0013	0.0002	0.0446
8.0	0.0599		0.0018	0.0004	0.0371
10	0.0510		0.0021	0.0006	0.0321
15	0.0377		0.0028	0.0011	0.0249
20	0.0302		0.0033	0.0015	0.0209
30	0.0220		0.0040	0.0021	0.0168
40	0.01746		0.0045	0.0026	0.0147
50	0.01456		0.0048	0.0029	0.0133
60	0.01254		0.0051	0.0033	0.0125
80	0.00988		0.0056	0.0038	0.0115
100	0.00820		0.0059	0.0042	0.0109

* Total scattering for hydrogen is given by the Klein-Nishina formula for free electrons.

† Barns/atom $\times 0.5997 = \text{cm}^2/\text{g}$.

Table 8-2. Beryllium

Photon energy, Mev	Scattering *		Photo-electric <i>K</i> and <i>L</i> shells, barns/atom	Pair production		Total †	
	With coherent, barns/atom	Without coherent, barns/atom		Nucleus, barns/atom	Electron, barns/atom	With coherent, cm ² /g	Without coherent, cm ² /g
0.01	3.54	2.56	5.42			0.599	0.533
0.015	3.01	2.52	1.39			0.294	0.261
0.02	2.77	2.47	0.52			0.220	0.200
0.03	2.53	2.39	0.13			0.178	0.168
0.04	2.38	2.31	0.052			0.163	0.158
0.05	2.29	2.24	0.021			0.154	0.151
0.06	2.21	2.18	0.010			0.148	0.146
0.08	2.10	2.07				0.140	0.138
0.10	1.99	1.972				0.133	0.132
0.15	1.78	1.774				0.119	0.119
0.20	1.63	1.626				0.109	0.109
0.30		1.414					0.0945
0.40		1.267					0.0847
0.50		1.157					0.0773
0.60		1.070					0.0715
0.80		0.940					0.0628
1.0		0.845					0.0565
1.5		0.686		0.00071			0.0459
2.0		0.586		0.0028			0.0394
3.0		0.460		0.0081	0.00005		0.0313
4.0		0.384		0.013	0.0002		0.0266
5.0		0.331		0.018	0.0004		0.0234
6.0		0.293		0.022	0.0008		0.0211
8.0		0.240		0.028	0.002		0.0180
10		0.204		0.034	0.003		0.0161
15		0.1509		0.044	0.004		0.0133
20		0.1210		0.052	0.006		0.0120 ‡
30		0.0880		0.063	0.008		0.0106
40		0.0698		0.070	0.010		0.0100
50		0.0582		0.076	0.012		0.00977
60		0.0502		0.081	0.013		0.00964
80		0.0395		0.087	0.015		0.00946
100		0.0328		0.093	0.017		0.00955

* Data in the first column are given by the sum of coherent scattering and of incoherent scattering from the Klein-Nishina formula corrected for binding effects. In the second column incoherent scattering is given by the Klein-Nishina formula for free electrons.

† Barns/atom $\times 0.06684 = \text{cm}^2/\text{g}$.

‡ Energy region in which dipole absorption attains a maximum cross section.

Table 8-3. Carbon

Photon energy, Mev	Scattering *		Photo-electric <i>K</i> and <i>L</i> shells, barns/atom	Pair production		Total †	
	With coherent, barns/atom	Without coherent, barns/atom		Nucleus, barns/atom	Electron, barns/atom	With coherent, cm ² /g	Without coherent, cm ² /g
0.01	6.88	3.84	38.6			2.28	2.13
0.015	5.30	3.77	10.2			0.777	0.701
0.02	4.64	3.71	3.91			0.429	0.382
0.03	4.04	3.58	0.99			0.252	0.229
0.04	3.71	3.47	0.38			0.205	0.193
0.05	3.50	3.37	0.18			0.185	0.178
0.06	3.37	3.28	0.096			0.174	0.169
0.08	3.18	3.10	0.037			0.161	0.157
0.10	3.02	2.96	0.017			0.152	0.149
0.15	2.69	2.66	0.0040			0.135	0.134
0.20	2.46	2.44				0.123	0.122
0.30	2.13	2.12				0.107	0.106
0.40		1.900					0.0953
0.50		1.735					0.0870
0.60		1.605					0.0805
0.80		1.410					0.0707
1.0		1.267					0.0636
1.5		1.030		0.0016			0.0518
2.0		0.878		0.0063			0.0444
3.0		0.691		0.018	0.00007		0.0366
4.0		0.576		0.030	0.0003		0.0304
5.0		0.497		0.040	0.0007		0.0270
6.0		0.439		0.048	0.001		0.0245
8.0		0.359		0.063	0.002		0.0213
10		0.306		0.076	0.004		0.0194
15		0.226		0.099	0.006		0.0166
20		0.1814		0.116	0.009		0.0154 ‡
30		0.1319		0.140	0.012		0.0142
40		0.1048		0.157	0.015		0.0139
50		0.0874		0.170	0.018		0.0138
60		0.0752		0.180	0.020		0.0138
80		0.0593		0.195	0.023		0.0139
100		0.0492		0.207	0.025		0.0141

* Data in the first column are given by the sum of coherent scattering and of incoherent scattering from the Klein-Nishina formula corrected for binding effects. In the second column incoherent scattering is given by the Klein-Nishina formula for free electrons.

† Barns/atom $\times 0.05016 = \text{cm}^2/\text{g}$.

‡ Energy region in which dipole absorption attains a maximum cross section.

Table 8-4. Nitrogen

Photon energy, Mev	Scattering *		Photo-electric <i>K</i> and <i>L</i> shells, barns/atom	Pair production		Total †	
	With coherent, barns/atom	Without coherent, barns/atom		Nucleus, barns/atom	Electron, barns/atom	With coherent, cm ² /g	Without coherent, cm ² /g
0.01	8.96	4.48	79.4			3.80	3.61
0.015	6.72	4.40	21.2			1.20	1.10
0.02	5.73	4.33	8.21			0.600	0.539
0.03	4.84	4.18	2.15			0.301	0.272
0.04	4.45	4.05	0.81			0.226	0.209
0.05	4.14	3.93	0.38			0.194	0.185
0.06	3.98	3.82	0.21			0.180	0.173
0.08	3.73	3.62	0.082			0.164	0.159
0.10	3.54	3.45	0.041			0.154	0.150
0.15	3.15	3.11	0.010			0.136	0.134
0.20	2.87	2.85				0.123	0.123
0.30	2.48	2.47				0.107	0.106
0.40		2.22					0.0955
0.50		2.02					0.0869
0.60		1.872					0.0805
0.80		1.645					0.0707
1.0		1.478					0.0636
1.5		1.201		0.0022			0.0517
2.0		1.025		0.0086			0.0445
3.0		0.806		0.025	0.00009		0.0357
4.0		0.672		0.040	0.0003		0.0306
5.0		0.580		0.054	0.0008		0.0273
6.0		0.512		0.066	0.001		0.0249
8.0		0.419		0.086	0.003		0.0218
10		0.357		0.103	0.004		0.0200
15		0.264		0.134	0.008		0.0175
20		0.212		0.158	0.010		0.0163
30		0.1539		0.190	0.015		0.0154 ‡
40		0.1222		0.213	0.018		0.0152
50		0.1019		0.231	0.020		0.0152
60		0.0878		0.244	0.023		0.0153
80		0.0692		0.264	0.026		0.0154
100		0.0574		0.280	0.029		0.0158

* Data in the first column are given by the sum of coherent scattering and of incoherent scattering from the Klein-Nishina formula corrected for binding effects. In the second column incoherent scattering is given by the Klein-Nishina formula for free electrons.

† Barns/atom $\times 0.04301 = \text{cm}^2/\text{g}$.

‡ Energy region in which dipole absorption attains a maximum cross section.

Table 8-5. Oxygen

Photon energy, Mev	Scattering *		Photo-electric K and L shells, barns/atom	Pair production		Total †	
	With coherent, barns/atom	Without coherent, barns/atom		Nucleus, barns/atom	Electron, barns/atom	With coherent, cm ² /g	Without coherent, cm ² /g
0.01	11.5	5.12	146			5.93	5.69
0.015	8.28	5.03	39.6			1.80	1.68
0.02	6.95	4.94	15.4			0.842	0.766
0.03	5.77	4.78	4.09			0.371	0.334
0.04	5.18	4.62	1.55			0.253	0.232
0.05	4.80	4.49	0.73			0.208	0.197
0.06	4.61	4.37	0.40			0.189	0.180
0.08	4.30	4.14	0.15			0.168	0.162
0.10	4.06	3.94	0.071			0.156	0.151
0.15	3.61	3.55	0.020			0.137	0.134
0.20	3.29	3.25	0.010			0.124	0.123
0.30	2.84	2.83				0.107	0.107
0.40	2.54	2.53				0.0956	0.0953
0.50		2.31					0.0870
0.60		2.14					0.0806
0.80		1.880					0.0708
1.0		1.690					0.0636
1.5		1.373		0.0028			0.0518
2.0		1.171		0.011			0.0445
3.0		0.921		0.032	0.0001		0.0359
4.0		0.768		0.053	0.0004		0.0309
5.0		0.663		0.070	0.0009		0.0276
6.0		0.586		0.086	0.002		0.0254
8.0		0.479		0.112	0.003		0.0224
10		0.408		0.134	0.005		0.0206
15		0.302		0.175	0.009		0.0183
20		0.242		0.206	0.012		0.0173 ‡
30		0.1759		0.248	0.017		0.0166
40		0.1397		0.278	0.021		0.0165
50		0.1165		0.300	0.023		0.0165
60		0.1003		0.317	0.026		0.0167
80		0.0790		0.344	0.030		0.0171
100		0.0656		0.364	0.034		0.0175

* Data in the first column are given by the sum of coherent scattering and of incoherent scattering from the Klein-Nishina formula corrected for binding effects. In the second column incoherent scattering is given by the Klein-Nishina formula for free electrons.

† Barns/atom $\times 0.03765 = \text{cm}^2/\text{g}$.

‡ Energy region in which dipole absorption attains a maximum cross section.

Table 8-6. Sodium

Photon energy, Mev	Scattering *		Photo-electric <i>K</i> , <i>L</i> , and <i>M</i> shells, barns/atom	Pair production		Total †	
	With coherent, barns/atom	Without coherent, barns/atom		Nucleus, barns/atom	Electron, barns/atom	With coherent, cm ² /g	Without coherent, cm ² /g
0.01	20	7.04	588			15.9	15.6
0.015	14	6.92	169			4.80	4.61
0.02	11.2	6.80	67.5			2.06	1.95
0.03	8.8	6.57	18.1			0.705	0.646
0.04	7.8	6.36	7.0			0.388	0.350
0.05	7.1	6.17	3.3			0.273	0.248
0.06	6.67	6.01	1.87			0.224	0.206
0.08	6.08	5.69	0.74			0.179	0.168
0.10	5.70	5.42	0.35			0.159	0.151
0.15	5.01	4.88	0.091			0.134	0.130
0.20	4.54	4.47	0.040			0.120	0.118
0.30	3.92	3.89	0.010			0.103	0.102
0.40	3.50	3.48				0.0917	0.0912
0.50	3.19	3.18				0.0836	0.0833
0.60		2.94					0.0770
0.80		2.58					0.0676
1.0		2.32					0.0608
1.5		1.888		0.0054			0.0496
2.0		1.610		0.021			0.0427
3.0		1.266		0.061	0.0001		0.0348
4.0		1.056		0.100	0.0005		0.0303
5.0		0.911		0.133	0.001		0.0274
6.0		0.805		0.163	0.002		0.0254
8.0		0.659		0.211	0.004		0.0229
10		0.561		0.252	0.007		0.0215
15		0.415		0.330	0.012		0.0198
20		0.333		0.387	0.016		0.0193
30		0.242		0.465	0.023		0.0191
40		0.1921		0.521	0.028		0.0194
50		0.1602		0.562	0.032		0.0198
60		0.1379		0.595	0.036		0.0201
80		0.1087		0.645	0.041		0.0208
100		0.0901		0.680	0.046		0.0214

* Data in the first column are given by the sum of coherent scattering and of incoherent scattering from the Klein-Nishina formula corrected for binding effects. In the second column incoherent scattering is given by the Klein-Nishina formula for free electrons.

† Barns/atom $\times 0.02620 = \text{cm}^2/\text{g}$.

Table 8-7. Magnesium

Photon energy, Mev	Scattering *		Photo-electric <i>K</i> , <i>L</i> , and <i>M</i> shells, barns/atom	Pair production		Total †	
	With coherent, barns/atom	Without coherent, barns/atom		Nucleus, barns/atom	Electron, barns/atom	With coherent, cm ² /g	Without coherent, cm ² /g
0.01	25	7.68	847			21.6	21.2
0.015	17	7.55	246			6.51	6.28
0.02	13	7.42	99.7			2.79	2.65
0.03	10.2	7.16	27.2			0.926	0.851
0.04	8.7	6.94	10.6			0.478	0.434
0.05	7.9	6.73	5.1			0.322	0.293
0.06	7.4	6.55	2.8			0.253	0.232
0.08	6.66	6.20	1.11			0.192	0.181
0.10	6.24	5.91	0.53			0.168	0.160
0.15	5.48	5.32	0.14			0.139	0.135
0.20	4.97	4.88	0.060			0.125	0.122
0.30	4.28	4.24	0.020			0.107	0.106
0.40	3.82	3.80	0.010			0.0949	0.0944
0.50	3.48	3.47				0.0862	0.0860
0.60		3.21					0.0795
0.80		2.82					0.0699
1.0		2.53					0.0627
1.5		2.06		0.0064			0.0512
2.0		1.757		0.026			0.0442
3.0		1.381		0.073	0.0001		0.0360
4.0		1.152		0.119	0.0006		0.0315
5.0		0.994		0.159	0.001		0.0286
6.0		0.878		0.194	0.002		0.0266
8.0		0.719		0.251	0.005		0.0242
10		0.612		0.300	0.007		0.0228
15		0.453		0.393	0.013		0.0213
20		0.363		0.459	0.018		0.0208
30		0.264		0.553	0.025		0.0209
40		0.210		0.619	0.031		0.0213
50		0.1747		0.667	0.035		0.0217
60		0.1505		0.707	0.039		0.0222
80		0.1185		0.765	0.045		0.0230
100		0.0983		0.807	0.050		0.0237

* Data in the first column are given by the sum of coherent scattering and of incoherent scattering from the Klein-Nishina formula corrected for binding effects. In the second column incoherent scattering is given by the Klein-Nishina formula for free electrons.

† Barns/atom $\times 0.02477 = \text{cm}^2/\text{g}$.

Table 8-8. Aluminum

Photon energy, Mev	Scattering *		Photo-electric <i>K</i> , <i>L</i> , and <i>M</i> shells, barns/atom	Pair production		Total †	
	With coherent, barns/atom	Without coherent, barns/atom		Nucleus, barns/atom	Electron, barns/atom	With coherent, cm ² /g	Without coherent, cm ² /g
0.01	29	8.32	1,170			26.8	26.3
0.015	19	8.18	343			8.08	7.84
0.02	15	8.03	141			3.48	3.33
0.03	11.5	7.76	39.0			1.13	1.04
0.04	9.8	7.51	15.2			0.558	0.507
0.05	8.8	7.29	7.3			0.360	0.326
0.06	8.1	7.10	4.0			0.270	0.248
0.08	7.26	6.72	1.61			0.198	0.186
0.10	6.79	6.41	0.78			0.169	0.161
0.15	5.96	5.77	0.21			0.138	0.134
0.20	5.39	5.29	0.080			0.122	0.120
0.30	4.64	4.60	0.020			0.104	0.103
0.40	4.14	4.12	0.010			0.0927	0.0922
0.50	3.78	3.76				0.0844	0.0840
0.60	3.49	3.48				0.0779	0.0777
0.80		3.06					0.0683
1.0		2.75					0.0614
1.5		2.23		0.0076			0.0500
2.0		1.903		0.030			0.0432
3.0		1.496		0.086	0.0002		0.0353
4.0		1.247		0.140	0.0006		0.0310
5.0		1.077		0.186	0.001		0.0282
6.0		0.952		0.227	0.002		0.0264
8.0		0.778		0.295	0.005		0.0241
10		0.663		0.353	0.008		0.0229
15		0.490		0.460	0.014		0.0215
20		0.393		0.539	0.019		0.0212 ‡
30		0.286		0.647	0.027		0.0214
40		0.227		0.726	0.033		0.0220
50		0.1893		0.782	0.038		0.0225
60		0.1630		0.828	0.042		0.0231
80		0.1284		0.896	0.049		0.0240
100		0.1065		0.944	0.055		0.0247

* Data in the first column are given by the sum of coherent scattering and of incoherent scattering from the Klein-Nishina formula corrected for binding effects. In the second column incoherent scattering is given by the Klein-Nishina formula for free electrons.

† Barns/atom $\times 0.02233 = \text{cm}^2/\text{g}$.

‡ Energy region in which dipole absorption attains a maximum cross section.

Table 8-9. Silicon

Photon energy, Mev	Scattering *		Photo-electric <i>K</i> , <i>L</i> , and <i>M</i> shells, barns/atom	Pair production		Total †	
	With coherent, barns/atom	Without coherent, barns/atom		Nucleus, barns/atom	Electron, barns/atom	With coherent, cm ² /g	Without coherent, cm ² /g
0.01	33	8.96	1,580			34.6	34.1
0.015	22	8.81	470			10.6	10.3
0.02	17	8.65	194			4.53	4.35
0.03	12.8	8.36	54.4			1.44	1.35
0.04	10.8	8.09	21.4			0.691	0.633
0.05	9.7	7.85	10.3			0.429	0.389
0.06	8.9	7.64	5.8			0.315	0.288
0.08	8.0	7.24	2.3			0.221	0.205
0.10	7.38	6.90	1.11			0.182	0.172
0.15	6.44	6.21	0.29			0.144	0.139
0.20	5.82	5.69	0.12			0.127	0.125
0.30	5.01	4.95	0.040			0.108	0.107
0.40	4.46	4.43	0.020			0.0961	0.0954
0.50	4.07	4.05				0.0873	0.0869
0.60	3.75	3.74				0.0804	0.0802
0.80	3.30	3.29				0.0708	0.0706
1.0		2.96					0.0635
1.5		2.40		0.0088			0.0517
2.0		2.05		0.035			0.0447
3.0		1.611		0.100	0.0002		0.0367
4.0		1.343		0.162	0.0007		0.0323
5.0		1.160		0.216	0.002		0.0296
6.0		1.025		0.264	0.003		0.0277
8.0		0.838		0.342	0.006		0.0254
10		0.714		0.408	0.009		0.0243
15		0.528		0.533	0.015		0.0231
20		0.423		0.623	0.021		0.0229 ‡
30		0.308		0.749	0.029		0.0233
40		0.244		0.838	0.036		0.0240
50		0.204		0.904	0.041		0.0246
60		0.1756		0.957	0.046		0.0253
80		0.1383		1.03	0.053		0.0262
100		0.1147		1.09	0.059		0.0271

* Data in the first column are given by the sum of coherent scattering and of incoherent scattering from the Klein-Nishina formula corrected for binding effects. In the second column incoherent scattering is given by the Klein-Nishina formula for free electrons.

† Barns/atom $\times 0.02145 = \text{cm}^2/\text{g}$.

‡ Energy region in which dipole absorption attains a maximum cross section.

Table 8-10. Phosphorus

Photon energy, Mev	Scattering *		Photo-electric <i>K, L, and M</i> shells, barns/atom	Pair production		Total †	
	With coherent, barns/atom	Without coherent, barns/atom		Nucleus, barns/atom	Electron, barns/atom	With coherent, cm ² /g	Without coherent, cm ² /g
0.01	38	9.60	2,090			41.4	40.8
0.015	25	9.44	619			12.5	12.2
0.02	19	9.27	259			5.41	5.22
0.03	14.3	8.96	74.3			1.72	1.62
0.04	12.0	8.67	28.8			0.794	0.729
0.05	10.6	8.42	13.8			0.475	0.432
0.06	9.7	8.19	7.8			0.340	0.311
0.08	8.6	7.76	3.1			0.228	0.211
0.10	7.98	7.39	1.55			0.185	0.174
0.15	6.93	6.65	0.40			0.143	0.137
0.20	6.26	6.10	0.17			0.125	0.122
0.30	5.37	5.30	0.05			0.105	0.104
0.40	4.79	4.75	0.02			0.0936	0.0928
0.50	4.36	4.34	0.01			0.0850	0.0846
0.60	4.02	4.01				0.0782	0.0780
0.80	3.53	3.52				0.0687	0.0685
1.0		3.17					0.0617
1.5		2.57		0.010			0.0502
2.0		2.20		0.040			0.0436
3.0		1.726		0.114	0.0002		0.0358
4.0		1.439		0.186	0.0007		0.0316
5.0		1.243		0.248	0.002		0.0290
6.0		1.098		0.302	0.003		0.0273
8.0		0.898		0.393	0.006		0.0252
10		0.765		0.469	0.009		0.0242
15		0.566		0.610	0.016		0.0232
20		0.454		0.714	0.022		0.0231
30		0.330		0.858	0.031		0.0237
40		0.262		0.961	0.038		0.0245
50		0.218		1.03	0.044		0.0251
60		0.1881		1.10	0.049		0.0260
80		0.1482		1.19	0.056		0.0271
100		0.1229		1.25	0.063		0.0279

* Data in the first column are given by the sum of coherent scattering and of incoherent scattering from the Klein-Nishina formula corrected for binding effects. In the second column incoherent scattering is given by the Klein-Nishina formula for free electrons.

† Barns/atom $\times 0.01945 = \text{cm}^2/\text{g}$.

Table 8-11. Sulfur

Photon energy, Mev	Scattering *		Photo-electric <i>K</i> , <i>L</i> , and <i>M</i> shells, barns/atom	Pair production		Total †	
	With coherent, barns/atom	Without coherent, barns/atom		Nucleus, barns/atom	Electron, barns/atom	With coherent, cm ² /g	Without coherent, cm ² /g
0.01	44	10.24	2.700			51.6	50.9
0.015	29	10.06	820			16.0	15.6
0.02	22	9.89	344			6.88	6.65
0.03	15.9	9.55	98.7			2.15	2.03
0.04	13.2	9.25	38.5			0.971	0.897
0.05	11.6	8.98	18.6			0.567	0.518
0.06	10.7	8.74	10.6			0.400	0.363
0.08	9.3	8.27	4.2			0.254	0.234
0.10	8.6	7.89	2.1			0.201	0.188
0.15	7.43	7.10	0.57			0.150	0.144
0.20	6.69	6.51	0.23			0.130	0.127
0.30	5.74	5.66	0.070			0.109	0.108
0.40	5.12	5.07	0.030			0.0968	0.0958
0.50	4.66	4.63	0.020			0.0879	0.0874
0.60	4.30	4.28	0.010			0.0810	0.0806
0.80	3.77	3.76				0.0708	0.0707
1.0	3.39	3.38				0.0637	0.0635
1.5		2.75		0.012			0.0519
2.0		2.34		0.046			0.0448
3.0		1.842		0.13	0.0002		0.0371
4.0		1.535		0.21	0.0008		0.0328
5.0		1.325		0.28	0.002		0.0302
6.0		1.171		0.34	0.003		0.0284
8.0		0.958		0.45	0.006		0.0266
10		0.816		0.53	0.010		0.0255
15		0.604		0.69	0.017		0.0246
20		0.484		0.81	0.023		0.0247
30		0.352		0.98	0.033		0.0256
40		0.279		1.09	0.041		0.0265
50		0.233		1.18	0.047		0.0274
60		0.201		1.24	0.052		0.0281
80		0.1580		1.34	0.060		0.0293
100		0.1311		1.42	0.067		0.0304

* Data in the first column are given by the sum of coherent scattering and of incoherent scattering from the Klein-Nishina formula corrected for binding effects. In the second column incoherent scattering is given by the Klein-Nishina formula for free electrons.

† Barns/atom $\times 0.01879 = \text{cm}^2/\text{g}$.

Table 8-12. Argon

Photon energy, Mev	Scattering *		Photo-electric <i>K, L, and M</i> shells, barns/atom	Pair production		Total †	
	With coherent, barns/atom	Without coherent, barns/atom		Nucleus, barns/atom	Electron, barns/atom	With coherent, cm ² /g	Without coherent, cm ² /g
0.01	56	11.52	4,280			65.4	64.7
0.015	36	11.32	1,320			20.5	20.1
0.02	28	11.12	561			8.88	8.63
0.03	19	10.75	164			2.76	2.64
0.04	15.8	10.40	64.5			1.21	1.13
0.05	13.6	10.10	31.6			0.682	0.629
0.06	12.4	9.83	18.0			0.459	0.420
0.08	10.8	9.31	7.2			0.271	0.249
0.10	9.85	8.87	3.6			0.203	0.188
0.15	8.43	7.98	0.98			0.142	0.135
0.20	7.57	7.32	0.41			0.120	0.117
0.30	6.48	6.36	0.12			0.0995	0.0977
0.40	5.76	5.70	0.050			0.0876	0.0867
0.50	5.24	5.21	0.030			0.0795	0.0790
0.60	4.84	4.82	0.020			0.0733	0.0730
0.80	4.24	4.23				0.0640	0.0638
1.0	3.81	3.80				0.0575	0.0573
1.5		3.09		0.015			0.0468
2.0		2.64		0.058			0.0407
3.0		2.07		0.17	0.0002		0.0338
4.0		1.727		0.27	0.0009		0.0301
5.0		1.491		0.36	0.002		0.0279
6.0		1.318		0.44	0.003		0.0266
8.0		1.078		0.56	0.007		0.0248
10		0.918		0.67	0.011		0.0241
15		0.679		0.87	0.019		0.0237
20		0.544		1.02	0.026		0.0240
30		0.396		1.23	0.037		0.0251
40		0.314		1.37	0.046		0.0261
50		0.262		1.48	0.053		0.0271
60		0.226		1.57	0.059		0.0280
80		0.1778		1.69	0.068		0.0292
100		0.1475		1.78	0.076		0.0302

* Data in the first column are given by the sum of coherent scattering and of incoherent scattering from the Klein-Nishina formula corrected for binding effects. In the second column incoherent scattering is given by the Klein-Nishina formula for free electrons.

† Barns/atom $\times 0.01508 = \text{cm}^2/\text{g}$.

Table 8-13. Potassium

Photon energy, Mev	Scattering *		Photo-electric <i>K</i> , <i>L</i> , and <i>M</i> shells, barns/atom	Pair production		Total †	
	With coherent, barns/atom	Without coherent, barns/atom		Nucleus, barns/atom	Electron, barns/atom	With coherent, cm ² /g	Without coherent, cm ² /g
0.01	63	12.16	5,260			82.0	81.2
0.015	40	11.95	1,650			26.0	25.6
0.02	31	11.74	698			11.2	10.9
0.03	21	11.34	206			3.50	3.35
0.04	17.1	10.98	81.5			1.52	1.43
0.05	14.7	10.66	40.1			0.844	0.782
0.06	13.3	10.37	23.0			0.559	0.514
0.08	11.6	9.82	9.2			0.321	0.293
0.10	10.5	9.37	4.6			0.233	0.215
0.15	8.95	8.43	1.27			0.157	0.149
0.20	8.02	7.73	0.52			0.132	0.127
0.30	6.85	6.72	0.15			0.108	0.106
0.40	6.09	6.02	0.070			0.0949	0.0938
0.50	5.53	5.49	0.040			0.0858	0.0852
0.60	5.11	5.08	0.020			0.0791	0.0786
0.80	4.48	4.46	0.010			0.0692	0.0689
1.0	4.02	4.01				0.0619	0.0618
1.5		3.26		0.017			0.0505
2.0		2.78		0.065			0.0438
3.0		2.19		0.18	0.0002		0.0365
4.0		1.823		0.30	0.0009		0.0327
5.0		1.574		0.40	0.002		0.0305
6.0		1.391		0.48	0.004		0.0289
8.0		1.138		0.63	0.008		0.0274
10		0.969		0.75	0.012		0.0267
15		0.717		0.97	0.020		0.0263
20		0.575		1.14	0.028		0.0269
30		0.418		1.37	0.040		0.0282
40		0.332		1.53	0.049		0.0294
50		0.277		1.65	0.056		0.0306
60		0.238		1.74	0.062		0.0314
80		0.1877		1.88	0.072		0.0330
100		0.1557		1.98	0.080		0.0341

* Data in the first column are given by the sum of coherent scattering and of incoherent scattering from the Klein-Nishina formula corrected for binding effects. In the second column incoherent scattering is given by the Klein-Nishina formula for free electrons.

† Barns/atom $\times 0.01541 = \text{cm}^2/\text{g}$.

Table 8-14. Calcium

Photon energy, Mev	Scattering *		Photo-electric <i>K</i> , <i>L</i> , and <i>M</i> shells, barns/atom	Pair production		Total †	
	With coherent, barns/atom	Without coherent, barns/atom		Nucleus, barns/atom	Electron, barns/atom	With coherent, cm ² /g	Without coherent, cm ² /g
0.01	69	12.80	6,380			96.9	96.1
0.015	44	12.58	2,010			30.9	30.4
0.02	33	12.36	859			13.4	13.1
0.03	23	11.94	254			4.16	4.00
0.04	18.5	11.56	102			1.81	1.71
0.05	15.8	11.22	50.6			0.998	0.929
0.06	14.3	10.92	28.8			0.648	0.597
0.08	12.3	10.34	11.6			0.359	0.330
0.10	11.1	9.86	6.0			0.257	0.238
0.15	9.48	8.87	1.63			0.167	0.158
0.20	8.47	8.13	0.67			0.137	0.132
0.30	7.23	7.07	0.20			0.112	0.109
0.40	6.42	6.33	0.090			0.0979	0.0965
0.50	5.84	5.78	0.050			0.0885	0.0876
0.60	5.38	5.35	0.030			0.0813	0.0809
0.80	4.72	4.70	0.010			0.0711	0.0708
1.0	4.24	4.22				0.0637	0.0634
1.5		3.43		0.018			0.0518
2.0		2.93		0.072			0.0451
3.0		2.30		0.20	0.0002		0.0376
4.0		1.919		0.33	0.0009		0.0338
5.0		1.657		0.44	0.002		0.0316
6.0		1.464		0.54	0.004		0.0302
8.0		1.198		0.69	0.008		0.0285
10		1.020		0.83	0.012		0.0280
15		0.755		1.08	0.022		0.0279
20		0.605		1.26	0.029		0.0285 ‡
30		0.440		1.51	0.042		0.0299
40		0.349		1.69	0.051		0.0314
50		0.291		1.82	0.059		0.0326
60		0.251		1.93	0.065		0.0338
80		0.198		2.08	0.075		0.0354
100		0.1639		2.19	0.084		0.0366

* Data in the first column are given by the sum of coherent scattering and of incoherent scattering from the Klein-Nishina formula corrected for binding effects. In the second column incoherent scattering is given by the Klein-Nishina formula for free electrons.

† Barns/atom $\times 0.01503 = \text{cm}^2/\text{g}$.

‡ Energy region in which dipole absorption attains a maximum cross section.

Table 8-15. Iron

Photon energy, Mev	Scattering *		Photo-electric <i>K</i> , <i>L</i> , and <i>M</i> shells, barns/atom	Pair production		Total †	
	With coherent, barns/atom	Without coherent, barns/atom		Nucleus, barns/atom	Electron, barns/atom	With coherent, cm ² /g	Without coherent, cm ² /g
0.01	120	16.64	16,500			179	178
0.015	75	16.35	5,380			58.8	58.2
0.02	55	16.07	2,380			26.3	25.8
0.03	37	15.52	729			8.26	8.03
0.04	29	15.03	308			3.64	3.48
0.05	24	14.59	155			1.93	1.83
0.06	20.7	14.20	91			1.20	1.13
0.08	17.2	13.44	38			0.595	0.555
0.10	15.4	12.82	19.1			0.372	0.344
0.15	12.8	11.53	5.4			0.196	0.183
0.20	11.3	10.57	2.23			0.146	0.138
0.30	9.50	9.19	0.66			0.110	0.106
0.40	8.42	8.23	0.29			0.0940	0.0919
0.50	7.63	7.52	0.16			0.0840	0.0828
0.60	7.03	6.96	0.10			0.0769	0.0762
0.80	6.15	6.11	0.05			0.0669	0.0664
1.0	5.52	5.49	0.03			0.0599	0.0595
1.5		4.46		0.032			0.0485
2.0		3.81		0.12			0.0424
3.0		2.99		0.35	0.0003		0.0360
4.0		2.50		0.56	0.001		0.0330
5.0		2.15		0.75	0.003		0.0313
6.0		1.903		0.91	0.005		0.0304
8.0		1.557		1.17	0.011		0.0295
10		1.326		1.39	0.016		0.0294
15		0.981		1.81	0.028		0.0304
20		0.786		2.10	0.038		0.0315 ‡
30		0.572		2.52	0.054		0.0339
40		0.454		2.81	0.067		0.0359
50		0.379		3.03	0.076		0.0376
60		0.326		3.21	0.085		0.0391
80		0.257		3.46	0.098		0.0412
100		0.213		3.64	0.11		0.0427

* Data in the first column are given by the sum of coherent scattering and of incoherent scattering from the Klein-Nishina formula corrected for binding effects. In the second column incoherent scattering is given by the Klein-Nishina formula for free electrons.

† Barns/atom $\times 0.01079 = \text{cm}^2/\text{g}$.

‡ Energy region in which dipole absorption attains a maximum cross section.

Table 8-12. Argon

Photon energy, Mev	Scattering *		Photo-electric <i>K, L, and M</i> shells, barns/atom	Pair production		Total †	
	With coherent, barns/atom	Without coherent, barns/atom		Nucleus, barns/atom	Electron, barns/atom	With coherent, cm ² /g	Without coherent, cm ² /g
0.01	56	11.52	4,280			65.4	64.7
0.015	36	11.32	1,320			20.5	20.1
0.02	28	11.12	561			8.88	8.63
0.03	19	10.75	164			2.76	2.64
0.04	15.8	10.40	64.5			1.21	1.13
0.05	13.6	10.10	31.6			0.682	0.629
0.06	12.4	9.83	18.0			0.459	0.420
0.08	10.8	9.31	7.2			0.271	0.249
0.10	9.85	8.87	3.6			0.203	0.188
0.15	8.43	7.98	0.98			0.142	0.135
0.20	7.57	7.32	0.41			0.120	0.117
0.30	6.48	6.36	0.12			0.0995	0.0977
0.40	5.76	5.70	0.050			0.0876	0.0867
0.50	5.24	5.21	0.030			0.0795	0.0790
0.60	4.84	4.82	0.020			0.0733	0.0730
0.80	4.24	4.23				0.0640	0.0638
1.0	3.81	3.80				0.0575	0.0573
1.5		3.09		0.015			0.0468
2.0		2.64		0.058			0.0407
3.0		2.07		0.17	0.0002		0.0338
4.0		1.727		0.27	0.0009		0.0301
5.0		1.491		0.36	0.002		0.0279
6.0		1.318		0.44	0.003		0.0266
8.0		1.078		0.56	0.007		0.0248
10		0.918		0.67	0.011		0.0241
15		0.679		0.87	0.019		0.0237
20		0.544		1.02	0.026		0.0240
30		0.396		1.23	0.037		0.0251
40		0.314		1.37	0.046		0.0261
50		0.262		1.48	0.053		0.0271
60		0.226		1.57	0.059		0.0280
80		0.1778		1.69	0.068		0.0292
100		0.1475		1.78	0.076		0.0302

* Data in the first column are given by the sum of coherent scattering and of incoherent scattering from the Klein-Nishina formula corrected for binding effects. In the second column incoherent scattering is given by the Klein-Nishina formula for free electrons.

† Barns/atom $\times 0.01508 = \text{cm}^2/\text{g}$.

Table 8-17. Molybdenum

Photon energy, Mev	Scattering *		Photo-electric <i>K, L, and M shells, barns/atom</i>	Pair production		Total †	
	With coherent, barns/atom	Without coherent, barns/atom		Nucleus, barns/atom	Electron, barns/atom	With coherent, cm ² /g	Without coherent, cm ² /g
0.01	340	26.9	11,400			73.7	71.8
0.015	220	26.4	3,480			23.2	22.0
0.0200 ‡	160	26.0	1,510			10.5	9.64
0.0200	160	26.0	13,000			82.6	81.8
0.03	98	25.1	4,260			27.4	26.9
0.04	71	24.3	1,920			12.5	12.2
0.05	56	23.6	1,030			6.82	6.62
0.06	46	22.9	620			4.18	4.04
0.08	36	21.7	274			1.95	1.86
0.10	30	20.7	144			1.09	1.03
0.15	23.2	18.63	43.4			0.418	0.389
0.20	19.8	17.08	18.7			0.242	0.225
0.30	16.1	14.85	5.8			0.138	0.130
0.40	14.0	13.30	2.6			0.104	0.0998
0.50	12.6	12.15	1.4			0.0879	0.0851
0.60	11.5	11.24	0.88			0.0777	0.0761
0.80	10.0	9.87	0.45			0.0656	0.0648
1.0	8.96	8.87	0.29			0.0581	0.0575
1.5	7.25	7.21	0.14	0.095		0.0470	0.0467
2.0		6.15	0.09	0.35			0.0414
3.0		4.83	0.06	0.93	0.0005		0.0365
4.0		4.03	0.04	1.49	0.002		0.0349
5.0		3.48	0.03	1.96	0.005		0.0344
6.0		3.08	0.023	2.36	0.008		0.0344
8.0		2.52	0.017	3.00	0.02		0.0349
10		2.14	0.013	3.53	0.03		0.0359
15		1.585		4.58	0.04		0.0390 ¶
20		1.270		5.32	0.06		0.0418
30		0.924		6.39	0.09		0.0465
40		0.733		7.11	0.11		0.0499
50		0.612		7.65	0.12		0.0526
60		0.527		8.08	0.14		0.0549
80		0.415		8.69	0.16		0.0582
100		0.344		9.15	0.18		0.0607

* Data in the first column are given by the sum of coherent scattering and of incoherent scattering from the Klein-Nishina formula corrected for binding effects. In the second column incoherent scattering is given by the Klein-Nishina formula for free electrons.

† Barns/atom $\times 0.006279 = \text{cm}^2/\text{g}$.

‡ *K* edge; at this and lower energies data for the *L* and *M* shells are given while at this and higher energies data for the *L*, *M*, and *K* shells are given.

¶ Energy region in which dipole absorption attains a maximum cross section.

Table 8-14. Calcium

Photon energy, Mev	Scattering *		Photo-electric <i>K</i> , <i>L</i> , and <i>M</i> shells, barns/atom	Pair production		Total †	
	With coherent, barns/atom	Without coherent, barns/atom		Nucleus, barns/atom	Electron, barns/atom	With coherent, cm ² /g	Without coherent, cm ² /g
0.01	69	12.80	6,380			96.9	96.1
0.015	44	12.58	2,010			30.9	30.4
0.02	33	12.36	859			13.4	13.1
0.03	23	11.94	254			4.16	4.00
0.04	18.5	11.56	102			1.81	1.71
0.05	15.8	11.22	50.6			0.998	0.929
0.06	14.3	10.92	28.8			0.648	0.597
0.08	12.3	10.34	11.6			0.359	0.330
0.10	11.1	9.86	6.0			0.257	0.238
0.15	9.48	8.87	1.63			0.167	0.158
0.20	8.47	8.13	0.67			0.137	0.132
0.30	7.23	7.07	0.20			0.112	0.109
0.40	6.42	6.33	0.090			0.0979	0.0965
0.50	5.84	5.78	0.050			0.0885	0.0876
0.60	5.38	5.35	0.030			0.0813	0.0809
0.80	4.72	4.70	0.010			0.0711	0.0708
1.0	4.24	4.22				0.0637	0.0634
1.5		3.43		0.018			0.0518
2.0		2.93		0.072			0.0451
3.0		2.30		0.20	0.0002		0.0376
4.0		1.919		0.33	0.0009		0.0338
5.0		1.657		0.44	0.002		0.0316
6.0		1.464		0.54	0.004		0.0302
8.0		1.198		0.69	0.008		0.0285
10		1.020		0.83	0.012		0.0280
15		0.755		1.08	0.022		0.0279
20		0.605		1.26	0.029		0.0285 ‡
30		0.440		1.51	0.042		0.0299
40		0.349		1.69	0.051		0.0314
50		0.291		1.82	0.059		0.0326
60		0.251		1.93	0.065		0.0338
80		0.198		2.08	0.075		0.0354
100		0.1639		2.19	0.084		0.0366

* Data in the first column are given by the sum of coherent scattering and of incoherent scattering from the Klein-Nishina formula corrected for binding effects. In the second column incoherent scattering is given by the Klein-Nishina formula for free electrons.

† Barns/atom $\times 0.01503 = \text{cm}^2/\text{g}$.

‡ Energy region in which dipole absorption attains a maximum cross section.

Table 8-19. Iodine

Photon energy, Mev	Scattering *		Photo-electric <i>K</i> , <i>L</i> , and <i>M</i> shells, barns/atom	Pair production		Total †	
	With coherent, barns/atom	Without coherent, barns/atom		Nucleus, barns/atom	Electron, barns/atom	With coherent, cm ² /g	Without coherent, cm ² /g
0.01	590	33.9	29,800			144	142
0.015	380	33.3	9,360			46.2	44.6
0.02	270	32.8	4,130			20.9	19.8
0.03	160	31.6	1,260			6.74	6.13
0.03323‡	150	31.3	933			5.14	4.58
0.03323‡	150	31.3	7,510			36.4	35.8
0.04	120	30.6	4,490			21.9	21.5
0.05	89	29.7	2,470			12.1	11.9
0.06	72	28.9	1,500			7.46	7.26
0.08	54	27.4	677			3.47	3.34
0.10	44	26.1	360			1.92	1.83
0.15	32	23.5	113			0.688	0.648
0.20	26.5	21.5	50			0.363	0.339
0.30	21.0	18.74	16.0			0.176	0.165
0.40	18.1	16.78	7.2			0.120	0.114
0.50	16.2	15.33	3.9			0.0954	0.0913
0.60	14.8	14.18	2.5			0.0821	0.0792
0.80	12.8	12.46	1.3			0.0669	0.0653
1.0	11.4	11.19	0.84			0.0581	0.0571
1.5	9.18	9.10	0.41	0.17		0.0463	0.0460
2.0	7.81	7.76	0.26	0.59		0.0411	0.0409
3.0		6.10	0.16	1.53	0.0006		0.0370
4.0		5.09	0.11	2.39	0.003		0.0360
5.0		4.39	0.08	3.12	0.006		0.0361
6.0		3.88	0.07	3.72	0.01		0.0365
8.0		3.17	0.05	4.70	0.02		0.0377
10		2.70	0.04	5.52	0.03		0.0394
15		2.00	0.03	7.12	0.06		0.0437 ¶
20		1.603	0.02	8.26	0.08		0.0473
30		1.165		9.92	0.11		0.0532
40		0.925		11.0	0.14		0.0573
50		0.772		11.9	0.16		0.0600
60		0.665		12.5	0.17		0.0633
80		0.524		13.5	0.20		0.0675
100		0.434		14.1	0.22		0.0700

* Data in the first column are given by the sum of coherent scattering and of incoherent scattering from the Klein-Nishina formula corrected for binding effects. In the second column incoherent scattering is given by the Klein-Nishina formula for free electrons.

† Barns/atom $\times 0.004747 = \text{cm}^2/\text{g}$.

‡ *K* edge; at this and lower energies data for the *L* and *M* shells are given while at this and higher energies data for the *L*, *M*, and *K* shells are given.

¶ Energy region in which dipole absorption attains a maximum cross section.

Table 8-20. Tungsten

Photon energy, Mev	Scattering *		Photo-electric <i>K</i> , <i>L</i> , and <i>M</i> shells, barns/atom	Pair production		Total †	
	With coherent, barns/atom	Without coherent, barns/atom		Nucleus, barns/atom	Electron, barns/atom	With coherent, cm ² /g	Without coherent, cm ² /g
0.01	1,300	47.4	17,700			62.2	58.1
0.01022 ‡	1,200	47.3	16,800			59.0	55.2
0.01212 ¶	1,000	46.9	64,700			215	212
0.015	840	46.5	36,000			121	118
0.02	590	45.7	16,000			64.3	52.6
0.03	350	44.2	5,040			17.7	16.7
0.04	240	42.8	2,220			8.06	7.41
0.05	180	41.5	1,160			4.39	3.94
0.06	145	40.4	674			2.68	2.34
0.06964 §	122	39.4	437			1.83	1.56
0.06964	122	39.4	3,230			11.0	10.7
0.08	104	38.3	2,250			7.71	7.49
0.10	80	36.5	1,250			4.36	4.21
0.15	54	32.8	408			1.51	1.44
0.20	42	30.1	186			0.747	0.708
0.30	31.5	26.2	63.1			0.310	0.293
0.40	26.5	23.4	29.8			0.184	0.174
0.50	23.4	21.4	16.7			0.131	0.125
0.60	21.2	19.80	11.0			0.105	0.101
0.80	18.2	17.39	5.9			0.0789	0.0763
1.0	16.1	15.63	3.9			0.0655	0.0640
1.5	12.9	12.70	1.9	0.41		0.0498	0.0492
2.0	10.9	10.83	1.2	1.32		0.0440	0.0437
3.0	8.57	8.52	0.71	3.13	0.0009	0.0407	0.0405
4.0		7.10	0.50	4.68	0.004		0.0402
5.0		6.13	0.38	5.96	0.008		0.0409
6.0		5.42	0.31	7.02	0.01		0.0418
8.0		4.43	0.23	8.68	0.03		0.0438
10		3.77	0.18	10.2	0.04		0.0465
15		2.79	0.11	13.1	0.08		0.0527
20		2.24	0.08	15.2	0.11		0.0578
30		1.627	0.06	18.3	0.15		0.0660
40		1.292	0.04	20.3	0.19		0.0715
50		1.077		21.8	0.22		0.0757
60		0.928		23.1	0.24		0.0795
80		0.731		24.8	0.28		0.0845
100		0.606		26.1	0.31		0.0885

* Data in the first column are given by the sum of coherent scattering and of incoherent scattering from the Klein-Nishina formula corrected for binding effects. In the second column incoherent scattering is given by the Klein-Nishina formula for free electrons.

† Barns/atom $\times 0.003276 = \text{cm}^2/\text{g}$.

‡ *L*₃ edge; at this and lower energies data for the *M* shell are given.

¶ *L*₁ edge; from this energy to the *K* edge energy data for the *L* and *M* shells are given.

§ *K* edge; at this and higher energies data for the *L*, *M*, and *K* shells are given.

Table 8-17. Molybdenum

Photon energy, Mev	Scattering *		Photo-electric <i>K</i> , <i>L</i> , and <i>M</i> shells, barns/atom	Pair production		Total †	
	With coherent, barns/atom	Without coherent, barns/atom		Nucleus, barns/atom	Electron, barns/atom	With coherent, cm ² /g	Without coherent, cm ² /g
0.01	340	26.9	11,400			73.7	71.8
0.015	220	26.4	3,480			23.2	22.0
0.0200 ‡	160	26.0	1,510			10.5	9.64
0.0200	160	26.0	13,000			82.6	81.8
0.03	98	25.1	4,260			27.4	26.9
0.04	71	24.3	1,920			12.5	12.2
0.05	56	23.6	1,030			6.82	6.62
0.06	46	22.9	620			4.18	4.04
0.08	36	21.7	274			1.95	1.86
0.10	30	20.7	144			1.09	1.03
0.15	23.2	18.63	43.4			0.418	0.389
0.20	19.8	17.08	18.7			0.242	0.225
0.30	16.1	14.85	5.8			0.138	0.130
0.40	14.0	13.30	2.6			0.104	0.0998
0.50	12.6	12.15	1.4			0.0879	0.0851
0.60	11.5	11.24	0.88			0.0777	0.0761
0.80	10.0	9.87	0.45			0.0656	0.0648
1.0	8.96	8.87	0.29			0.0581	0.0575
1.5	7.25	7.21	0.14	0.095		0.0470	0.0467
2.0		6.15	0.09	0.35			0.0414
3.0		4.83	0.05	0.93	0.0005		0.0365
4.0		4.03	0.04	1.49	0.002		0.0349
5.0		3.48	0.03	1.96	0.005		0.0344
6.0		3.08	0.023	2.36	0.008		0.0344
8.0		2.52	0.017	3.00	0.02		0.0349
10		2.14	0.013	3.53	0.03		0.0359
15		1.585		4.58	0.04		0.0390 ¶
20		1.270		5.32	0.06		0.0418
30		0.924		6.39	0.09		0.0465
40		0.733		7.11	0.11		0.0499
50		0.612		7.65	0.12		0.0526
60		0.527		8.08	0.14		0.0549
80		0.415		8.69	0.16		0.0582
100		0.344		9.15	0.18		0.0607

* Data in the first column are given by the sum of coherent scattering and of incoherent scattering from the Klein-Nishina formula corrected for binding effects. In the second column incoherent scattering is given by the Klein-Nishina formula for free electrons.

† Barns/atom $\times 0.006279$ = cm²/g.

‡ *K* edge; at this and lower energies data for the *L* and *M* shells are given while at this and higher energies data for the *L*, *M*, and *K* shells are given.

¶ Energy region in which dipole absorption attains a maximum cross section.

Table 8-18. Tin

Photon energy, Mev	Scattering *		Photo-electric <i>K</i> , <i>L</i> , and <i>M</i> shells, barns/atom	Pair production		Total †	
	With coherent, barns/atom	Without coherent, barns/atom		Nucleus, barns/atom	Electron, barns/atom	With coherent, cm ² /g	Without coherent, cm ² /g
0.01	510	32.0	24,000			124	122
0.015	340	31.4	7,410			39.3	37.8
0.02	240	30.9	3,220			17.6	16.5
0.02925 ‡	150	30.0	1,050			6.09	5.48
0.02925	150	30.0	8,580			44.3	43.7
0.03	140	29.8	8,150			42.1	41.5
0.04	100	28.9	3,700			19.3	18.9
0.05	79	28.0	1,990			10.5	10.2
0.06	65	27.3	1,210			6.47	6.28
0.08	49	25.8	539			2.98	2.87
0.10	40	24.6	286			1.65	1.58
0.15	29.6	22.2	88.8			0.601	0.563
0.20	24.6	20.3	39.3			0.324	0.303
0.30	19.7	17.68	12.4			0.163	0.153
0.40	17.0	15.84	5.6			0.115	0.109
0.50	15.2	14.46	3.0			0.0924	0.0886
0.60	13.8	13.38	1.9			0.0797	0.0776
0.80	12.0	11.75	1.0			0.0660	0.0647
1.0	10.7	10.56	0.64			0.0576	0.0568
1.5	8.65	8.58	0.32	0.14		0.0462	0.0459
2.0	7.36	7.32	0.20	0.51		0.0410	0.0408
3.0		5.76	0.12	1.35	0.0006		0.0367
4.0		4.80	0.08	2.12	0.002		0.0355
5.0		4.14	0.06	2.78	0.006		0.0355
6.0		3.66	0.05	3.33	0.01		0.0358
8.0		2.99	0.04	4.20	0.02		0.0368
10		2.55	0.03	4.94	0.03		0.0383
15		1.886	0.02	6.39	0.05		0.0424
20		1.512	0.015	7.40	0.07		0.0457
30		1.100		8.91	0.10		0.0513
40		0.873		9.89	0.13		0.0553
50		0.728		10.6	0.15		0.0533
60		0.627		11.2	0.16		0.0609
80		0.494		12.1	0.19		0.0649
100		0.410		12.7	0.21		0.0676

* Data in the first column are given by the sum of coherent scattering and of incoherent scattering from the Klein-Nishina formula corrected for binding effects. In the second column incoherent scattering is given by the Klein-Nishina formula for free electrons.

† Barns/atom $\times 0.005076 = \text{cm}^2/\text{g}$.

‡ *K* edge; at this and lower energies data for the *L* and *M* shells are given while at this and higher energies data for the *L*, *M*, and *K* shells are given.

Table 8-19. Iodine

Photon energy, Mev	Scattering *		Photo-electric <i>K</i> , <i>L</i> , and <i>M</i> shells, barns/atom	Pair production		Total †	
	With coherent, barns/atom	Without coherent, barns/atom		Nucleus, barns/atom	Electron, barns/atom	With coherent, cm ² /g	Without coherent, cm ² /g
0.01	590	33.9	29,800			144	142
0.015	380	33.3	9,360			46.2	44.6
0.02	270	32.8	4,130			20.9	19.8
0.03	160	31.6	1,260			6.74	6.13
0.03323‡	150	31.3	933			5.14	4.58
0.03323‡	150	31.3	7,510			36.4	35.8
0.04	120	30.6	4,490			21.9	21.5
0.05	89	29.7	2,470			12.1	11.9
0.06	72	28.9	1,500			7.46	7.26
0.08	54	27.4	677			3.47	3.34
0.10	44	26.1	360			1.92	1.83
0.15	32	23.5	113			0.688	0.648
0.20	26.5	21.5	50			0.363	0.339
0.30	21.0	18.74	16.0			0.176	0.165
0.40	18.1	16.78	7.2			0.120	0.114
0.50	16.2	15.33	3.9			0.0954	0.0913
0.60	14.8	14.18	2.5			0.0821	0.0792
0.80	12.8	12.46	1.3			0.0669	0.0653
1.0	11.4	11.19	0.84			0.0581	0.0571
1.5	9.18	9.10	0.41	0.17		0.0463	0.0460
2.0	7.81	7.76	0.26	0.59		0.0411	0.0409
3.0		6.10	0.16	1.53	0.0006		0.0370
4.0		5.09	0.11	2.39	0.003		0.0360
5.0		4.39	0.08	3.12	0.006		0.0361
6.0		3.88	0.07	3.72	0.01		0.0365
8.0		3.17	0.05	4.70	0.02		0.0377
10		2.70	0.04	5.52	0.03		0.0394
15		2.00	0.03	7.12	0.06		0.0437 ¶
20		1.603	0.02	8.26	0.08		0.0473
30		1.165		9.92	0.11		0.0532
40		0.925		11.0	0.14		0.0573
50		0.772		11.9	0.16		0.0600
60		0.665		12.5	0.17		0.0633
80		0.524		13.5	0.20		0.0675
100		0.434		14.1	0.22		0.0700

* Data in the first column are given by the sum of coherent scattering and of incoherent scattering from the Klein-Nishina formula corrected for binding effects. In the second column incoherent scattering is given by the Klein-Nishina formula for free electrons.

† Barns/atom $\times 0.004747 = \text{cm}^2/\text{g}$.

‡ *K* edge; at this and lower energies data for the *L* and *M* shells are given while at this and higher energies data for the *L*, *M*, and *K* shells are given.

¶ Energy region in which dipole absorption attains a maximum cross section.

Table 8-20. Tungsten

Photon energy, Mev	Scattering *		Photo-electric <i>K</i> , <i>L</i> , and <i>M</i> shells, barns/atom	Pair production		Total †	
	With coherent, barns/atom	Without coherent, barns/atom		Nucleus, barns/atom	Electron, barns/atom	With coherent, cm ² /g	Without coherent, cm ² /g
0.01	1,300	47.4	17,700			62.2	58.1
0.01022 ‡	1,200	47.3	16,800			59.0	55.2
0.01212 ¶	1,000	46.9	64,700			215	212
0.015	840	46.5	36,000			121	118
0.02	590	45.7	16,000			54.3	52.6
0.03	350	44.2	5,040			17.7	16.7
0.04	240	42.8	2,220			8.06	7.41
0.05	180	41.5	1,160			4.39	3.94
0.06	145	40.4	674			2.68	2.34
0.06964 §	122	39.4	437			1.83	1.56
0.06964	122	39.4	3,230			11.0	10.7
0.08	104	38.3	2,250			7.71	7.49
0.10	80	36.5	1,250			4.36	4.21
0.15	54	32.8	408			1.51	1.44
0.20	42	30.1	186			0.747	0.708
0.30	31.5	26.2	63.1			0.310	0.293
0.40	26.5	23.4	29.8			0.184	0.174
0.50	23.4	21.4	16.7			0.131	0.125
0.60	21.2	19.80	11.0			0.105	0.101
0.80	18.2	17.39	5.9			0.0789	0.0763
1.0	16.1	15.63	3.9			0.0655	0.0640
1.5	12.9	12.70	1.9	0.41		0.0498	0.0492
2.0	10.9	10.83	1.2	1.32		0.0440	0.0437
3.0	8.57	8.52	0.71	3.13	0.0009	0.0407	0.0405
4.0		7.10	0.50	4.68	0.004		0.0402
5.0		6.13	0.38	5.96	0.008		0.0409
6.0		5.42	0.31	7.02	0.01		0.0418
8.0		4.43	0.23	8.68	0.03		0.0438
10		3.77	0.18	10.2	0.04		0.0465
15		2.79	0.11	13.1	0.08		0.0527
20		2.24	0.08	15.2	0.11		0.0578
30		1.627	0.06	18.3	0.15		0.0660
40		1.292	0.04	20.3	0.19		0.0715
50		1.077		21.8	0.22		0.0757
60		0.928		23.1	0.24		0.0795
80		0.731		24.8	0.28		0.0845
100		0.606		26.1	0.31		0.0885

* Data in the first column are given by the sum of coherent scattering and of incoherent scattering from the Klein-Nishina formula corrected for binding effects. In the second column incoherent scattering is given by the Klein-Nishina formula for free electrons.

† Barns/atom $\times 0.003276 = \text{cm}^2/\text{g}$.

‡ *L*₃ edge; at this and lower energies data for the *M* shell are given.

¶ *L*₁ edge; from this energy to the *K* edge energy data for the *L* and *M* shells are given.

§ *K* edge; at this and higher energies data for the *L*, *M*, and *K* shells are given.

Table 8-21. Platinum

Photon energy, Mev	Scattering *		Photo-electric <i>K</i> , <i>L</i> , and <i>M</i> shells, barns/atom	Pair production		Total †	
	With coherent, barns/atom	Without coherent, barns/atom		Nucleus, barns/atom	Electron, barns/atom	With coherent, cm ² /g	Without coherent, cm ² /g
0.01	1,400	49.9	22,000			72.2	68.0
0.01158 ‡	1,200	49.6	14,800			49.4	45.8
0.01391 ¶	1,000	49.2	53,900			169	166
0.015	940	49.1	43,800			138	135
0.02	670	48.2	19,700			62.9	60.9
0.03	400	46.6	6,240			20.5	19.4
0.04	280	45.1	2,720			9.26	8.53
0.05	210	43.8	1,440			5.09	4.58
0.06	163	42.6	836			3.08	2.71
0.07858 §	117	40.6	380			1.53	1.30
0.07858 §	117	40.6	2,860			9.19	8.95
0.08	115	40.3	2,750			8.84	8.61
0.10	88	38.4	1,500			4.90	4.75
0.15	59	34.6	498			1.72	1.64
0.20	45	31.7	226			0.836	0.795
0.30	34	27.6	77.3			0.343	0.324
0.40	28.3	24.7	37.1			0.202	0.191
0.50	24.8	22.6	21.2			0.142	0.135
0.60	22.5	20.9	13.9			0.112	0.107
0.80	19.2	18.33	7.6			0.0827	0.0800
1.0	17.0	16.47	4.9			0.0676	0.0659
1.5	13.6	13.38	2.4	0.47		0.0508	0.0501
2.0	11.6	11.42	1.5	1.51		0.0451	0.0445
3.0	9.04	8.98	0.90	3.52	0.001	0.0415	0.0414
4.0	7.52	7.48	0.63	5.21	0.004	0.0412	0.0411
5.0		6.46	0.48	6.59	0.009		0.0418
6.0		5.71	0.39	7.73	0.02		0.0427
8.0		4.67	0.29	9.54	0.03		0.0448
10		3.98	0.22	11.2	0.05		0.0477
15		2.94	0.14	14.4	0.08		0.0542
20		2.36	0.10	16.7	0.11		0.0595
30		1.715	0.07	20.1	0.16		0.0680
40		1.362	0.06	22.3	0.20		0.0738
50		1.136	0.04	24.0	0.23		0.0784
60		0.978		25.4	0.25		0.0822
80		0.770		27.3	0.29		0.0875
100		0.639		28.6	0.33		0.0913

* Data in the first column are given by the sum of coherent scattering and of incoherent scattering from the Klein-Nishina formula corrected for binding effects. In the second column incoherent scattering is given by the Klein-Nishina formula for free electrons.

† Barns/atom $\times 0.003086 = \text{cm}^2/\text{g}$.

‡ *L*₃ edge; at this and lower energies data for the *M* shell are given.

¶ *L*₁ edge; from this energy to the *K* edge energy data for the *L* and *M* shells are given.

§ *K* edge; at this and higher energies data for the *L*, *M*, and *K* shells are given.

Table 8-22. Thallium

Photon energy, Mev	Scattering *		Photo-electric <i>K</i> , <i>L</i> , and <i>M</i> shells, barns/atom	Pair production		Total †	
	With coherent, barns/atom	Without coherent, barns/atom		Nucleus, barns/atom	Electron, barns/atom	With coherent, cm ² /g	Without coherent, cm ² /g
0.01	1,500	51.8	26,000			81.1	76.8
0.01268 ‡	1,200	51.3	13,400			43.0	39.7
0.01537 ¶	990	50.7	47,200			142	139
0.02	730	50.1	22,700			69.1	67.1
0.03	430	48.4	7,220			22.6	21.4
0.04	300	46.8	3,200			10.3	9.57
0.05	220	45.4	1,660			5.54	5.03
0.06	180	44.2	976			3.41	3.01
0.08	124	41.9	420			1.60	1.36
0.08584 §	114	41.3	341			1.34	1.13
0.08584 §	114	41.3	2,577			7.93	7.72
0.10	95	39.9	1,710			5.32	5.16
0.15	63	35.9	576			1.88	1.80
0.20	48	32.9	261			0.911	0.866
0.30	35.5	28.6	88.9			0.367	0.346
0.40	29.6	25.6	43.6			0.216	0.204
0.50	26.0	23.4	25.0			0.150	0.143
0.60	23.4	21.7	16.4			0.117	0.112
0.80	20.0	19.04	8.9			0.0852	0.0824
1.0	17.8	17.11	5.8			0.0696	0.0675
1.5	14.2	13.90	2.8	0.53		0.0517	0.0508
2.0	12.0	11.86	1.8	1.67		0.0456	0.0452
3.0	9.40	9.32	1.1	3.83	0.001	0.0422	0.0420
4.0	7.81	7.77	0.72	5.62	0.004	0.0417	0.0416
5.0		6.71	0.56	7.08	0.009		0.0423
6.0		5.93	0.45	8.29	0.02		0.0433
8.0		4.85	0.32	10.2	0.03		0.0454
10		4.13	0.25	12.0	0.05		0.0484
15		3.06	0.17	15.4	0.09		0.0552
20		2.45	0.12	17.9	0.12		0.0607
30		1.781	0.09	21.5	0.17		0.0694
40		1.414	0.07	23.9	0.21		0.0754
50		1.179	0.05	25.7	0.24		0.0801
60		1.016		27.1	0.26		0.0837
80		0.800		29.2	0.31		0.0894
100		0.664		30.6	0.34		0.0932

* Data in the first column are given by the sum of coherent scattering and of incoherent scattering from the Klein-Nishina formula corrected for binding effects. In the second column incoherent scattering is given by the Klein-Nishina formula for free electrons.

† Barns/atom $\times 0.002948 = \text{cm}^2/\text{g}$.

‡ *L*₁ edge; at this and lower energies data for the *M* shell are given.

¶ *L*₁ edge; from this energy to the *K* edge energy data for the *L* and *M* shells are given.

§ *K* edge; at this and higher energies data for the *L*, *M*, and *K* shells are given.

Table 8-23. Lead

Photon energy, Mev	Scattering *		Photo-electric <i>K, L, and M</i> shells, barns/atom	Pair production		Total †	
	With coherent, barns/atom	Without coherent, barns/atom		Nucleus, barns/atom	Electron, barns/atom	With coherent, cm ² /g	Without coherent, cm ² /g
0.01	1,600	52.5	27,500			84.6	80.1
0.01307 ‡	1,200	51.8	13,200			41.9	38.5
0.01589 ¶	980	51.3	45,400			135	132
0.02	750	50.7	24,000			72.0	69.9
0.03	450	49.0	7,620			23.5	22.3
0.04	310	47.4	3,310			10.5	9.76
0.05	230	46.0	1,740			5.73	5.19
0.06	180	44.8	1,040			3.55	3.15
0.08	127	42.4	444			1.66	1.41
0.08823 §	113	41.6	334			1.30	1.09
0.08823 §	113	41.6	2,510			7.63	7.42
0.10	100	40.4	1,780			5.47	5.29
0.15	64	36.4	596			1.92	1.84
0.20	49	33.3	275			0.942	0.896
0.30	36.2	29.0	93.4			0.377	0.356
0.40	30.1	26.0	45.7			0.220	0.208
0.50	26.3	23.7	26.1			0.152	0.145
0.60	23.8	21.9	17.3			0.119	0.114
0.80	20.3	19.27	9.5			0.0866	0.0836
1.0	18.0	17.32	6.2			0.0704	0.0684
1.5	14.4	14.07	3.0	0.55		0.0522	0.0512
2.0	12.2	12.00	2.0	1.72		0.0463	0.0457
3.0	9.51	9.44	1.1	3.93	0.001	0.0423	0.0421
4.0	7.91	7.87	0.80	5.76	0.004	0.0421	0.0420
5.0		6.79	0.60	7.25	0.009		0.0426
6.0		6.00	0.49	8.47	0.02		0.0436
8.0		4.91	0.35	10.5	0.03		0.0459
10		4.18	0.28	12.3	0.05		0.0489
15		3.09	0.18	15.7	0.09		0.0554 **
20		2.48	0.13	18.3	0.12		0.0611
30		1.803	0.09	21.9	0.17		0.0697
40		1.432	0.07	24.4	0.21		0.0759
50		1.194	0.05	26.2	0.24		0.0805
60		1.028		27.7	0.27		0.0843
80		0.810		29.8	0.31		0.0899
100		0.672		31.3	0.34		0.0939

* Data in the first column are given by the sum of coherent scattering and of incoherent scattering from the Klein-Nishina formula corrected for binding effects. In the second column incoherent scattering is given by the Klein-Nishina formula for free electrons.

† Barns/atom $\times 0.002908 = \text{cm}^2/\text{g}$.

‡ L_1 edge; at this and lower energies data for the M shell are given.

¶ L_1 edge; from this energy to the K edge energy data for the L and M shells are given.

§ K edge; at this and higher energies data for the L , M , and K shells are given.

** Energy region in which dipole absorption attains a maximum cross section.

Table 8-24. Uranium

Photon energy, Mev	Scattering *		Photo-electric <i>K</i> , <i>L</i> , and <i>M</i> shells, barns/atom	Pair production		Total †	
	With coherent, barns/atom	Without coherent, barns/atom		Nucleus, barns/atom	Electron, barns/atom	With coherent, cm ² /g	Without coherent, cm ² /g
0.01	2,100	58.9	44,600			118	113
0.015	1,400	57.9	14,500			40.2	36.8
0.01720 ‡	1,200	57.4	10,000			28.3	25.5
0.02181 ¶	880	56.5	29,400			76.6	74.6
0.03	590	54.9	12,000			31.9	30.5
0.04	400	53.2	5,250			14.3	13.4
0.05	300	51.6	2,780			7.79	7.17
0.06	230	50.2	1,640			4.73	4.28
0.08	163	47.6	716			2.22	1.93
0.10	123	45.3	374			1.26	1.06
0.1163 §	103	43.8	239			0.865	0.716
0.1163 §	103	43.8	1,790			4.79	4.64
0.15	78	40.8	916			2.52	2.42
0.20	59	37.4	425			1.22	1.17
0.30	42	32.5	146			0.476	0.452
0.40	34.7	29.1	73.2			0.273	0.259
0.50	30.2	26.6	43.1			0.185	0.176
0.60	27.1	24.6	29.2			0.142	0.136
0.80	23.0	21.6	16.0			0.0987	0.0952
1.0	20.3	19.43	10.5			0.0779	0.0757
1.5	16.2	15.79	5.1	0.77		0.0559	0.0548
2.0	13.7	13.47	3.3	2.35		0.0490	0.0484
3.0	10.7	10.59	1.9	5.09	0.001	0.0448	0.0445
4.0	8.88	8.83	1.3	7.26	0.004	0.0441	0.0440
5.0		7.62	1.0	9.00	0.01		0.0446
6.0		6.74	0.81	10.4	0.02		0.0455
8.0		5.51	0.59	12.8	0.04		0.0479
10		4.69	0.46	15.0	0.06		0.0511
15		3.47	0.30	19.3	0.10		0.0586 **
20		2.78	0.22	22.4	0.13		0.0646
30		2.023	0.15	26.8	0.19		0.0738
40		1.606	0.11	29.8	0.24		0.0804
50		1.340	0.09	32.1	0.27		0.0855
60		1.154		33.9	0.30		0.0895
80		0.909		36.5	0.35		0.0956
100		0.754		38.3	0.39		0.0998

* Data in the first column are given by the sum of coherent scattering and of incoherent scattering from the Klein-Nishina formula corrected for binding effects. In the second column incoherent scattering is given by the Klein-Nishina formula for free electrons.

† Barns/atom $\times 0.002531 = \text{cm}^2/\text{g}$.

‡ *L*₁ edge; at this and lower energies only *M*-shell data are given.

¶ *L*₁ edge; from this to the *K* edge energy data for the *L* and *M* shells are given.

§ *K* edge; at this and higher energies data for the *L*, *M*, and *K* shells are given.

** Energy region in which dipole absorption attains a maximum cross section.

Table 8-25. Water

Photon energy, Mev	Scattering *		Photo-electric <i>K</i> and <i>L</i> shells, barns/molecule	Pair production		Total †	
	With coherent, barns/molecule	Without coherent, barns/molecule		Nucleus, barns/molecule	Electron, barns/molecule	With coherent, cm ² /g	Without coherent, cm ² /g
0.01	12.8	6.40	146			5.31	5.10
0.015	9.54	6.29	39.6			1.64	1.53
0.02	8.19	6.18	15.4			0.789	0.722
0.03	6.96	5.97	4.09			0.370	0.336
0.04	6.34	5.78	1.55			0.264	0.245
0.05	5.92	5.61	0.73			0.222	0.212
0.06	5.70	5.46	0.40			0.204	0.196
0.08	5.33	5.17	0.15			0.183	0.178
0.10	5.05	4.93	0.071			0.171	0.167
0.15	4.50	4.44	0.020			0.151	0.149
0.20	4.10	4.07	0.010			0.137	0.136
0.30	3.55	3.54				0.119	0.118
0.40		3.17					0.106
0.50		2.89					0.0966
0.60		2.68					0.0896
0.80		2.35					0.0786
1.0		2.11					0.0706
1.5		1.716		0.0029			0.0575
2.0		1.464		0.011			0.0493
3.0		1.151		0.033	0.0001		0.0396
4.0		0.960		0.055	0.0004		0.0339
5.0		0.828		0.072	0.001		0.0301
6.0		0.732		0.089	0.002		0.0275
8.0		0.599		0.116	0.003		0.0240
10		0.510		0.138	0.006		0.0219
15		0.377		0.181	0.010		0.0190
20		0.302		0.213	0.014		0.0177
30		0.220		0.256	0.019		0.0166
40		0.1746		0.287	0.024		0.0162
50		0.1456		0.310	0.026		0.0161
60		0.1254		0.327	0.029		0.0161
80		0.0988		0.355	0.034		0.0163
100		0.0820		0.376	0.038		0.0166

* Data in the first column are given by the sum of coherent scattering and of incoherent scattering from the Klein-Nishina formula corrected for binding effects. In the second column incoherent scattering is given by the Klein-Nishina formula for free electrons.

† Barns/molecule $\times 0.03344 = \text{cm}^2/\text{g}$.

Table 8-26. Sodium Iodide

Photon energy, Mev	Scattering *		Photo-electric <i>K</i> , <i>L</i> , and <i>M</i> shells, barns/ molecule	Pair production		Total †	
	With coherent, barns/molecule	Without coherent, barns/molecule		Nucleus, barns/molecule	Electron, barns/molecule	With coherent, cm ² /g	Without coherent, cm ² /g
0.01	610	41.0	30,400			125	122
0.015	390	40.3	9,530			39.9	38.5
0.02	280	39.6	4,200			18.0	17.0
0.03	170	38.2	1,280			5.83	5.30
0.03323 ‡	160	37.8	946			4.45	3.95
0.03323	160	37.8	7,520			30.9	30.4
0.04	130	37.0	4,500			18.6	18.2
0.05	96	35.9	2,470			10.3	10.1
0.06	79	34.9	1,500			6.35	6.17
0.08	60	33.1	678			2.97	2.86
0.10	50	31.5	360			1.65	1.57
0.15	37	28.4	113			0.603	0.568
0.20	31	26.0	50.0			0.326	0.305
0.30	24.9	22.6	16.0			0.164	0.155
0.40	21.6	20.3	7.2			0.116	0.111
0.50	19.4	18.51	3.9			0.0936	0.0901
0.60	17.7	17.12	2.5			0.0812	0.0789
0.80	15.4	15.04	1.3			0.0671	0.0657
1.0	13.7	13.52	0.84			0.0584	0.0577
1.5	11.1	10.98	0.41	0.18		0.0470	0.0465
2.0	9.42	9.37	0.26	0.61		0.0414	0.0412
3.0		7.37	0.16	1.59	0.0007		0.0367
4.0		6.14	0.11	2.49	0.004		0.0351
5.0		5.30	0.08	3.25	0.007		0.0347
6.0		4.68	0.07	3.88	0.01		0.0347
8.0		3.83	0.05	4.91	0.02		0.0354
10		3.26	0.04	5.77	0.04		0.0366
15		2.41	0.03	7.45	0.07		0.0400
20		1.935	0.02	8.65	0.10		0.0430
30		1.407		10.4	0.13		0.0480
40		1.117		11.5	0.17		0.0514
50		0.932		12.5	0.19		0.0547
60		0.803		13.1	0.21		0.0567
80		0.632		14.1	0.24		0.0602
100		0.525		14.8	0.27		0.0627

* Data in the first column are given by the sum of coherent scattering and of incoherent scattering from the Klein-Nishina formula corrected for binding effects. In the second column incoherent scattering is given by the Klein-Nishina formula for free electrons.

† Barns/molecule $\times 0.004019 = \text{cm}^2/\text{g}$.

‡ *K* edge of iodine; at this and lower energies data for the *L* and *M* shells are given while at this and higher energies data for the *L*, *M*, and *K* shells are given.

Table 8-27. Calcium Phosphate

Photon energy, Mev	Scattering *		Photo-electric <i>K, L, and M</i> shells, barns/molecule	Pair production		Total †	
	With coherent, barns/molecule	Without coherent, barns/molecule		Nucleus, barns/molecule	Electron, barns/molecule	With coherent, cm ² /g	Without coherent, cm ² /g
0.01	375	98.6	24,500			48.3	47.8
0.015	248	96.9	7,580			15.2	14.9
0.02	193	95.2	3,220			6.63	6.44
0.03	144	91.9	943			2.11	2.01
0.04	121	89.0	376			0.965	0.903
0.05	107	86.4	185			0.567	0.527
0.06	99.2	84.1	105			0.397	0.367
0.08	88.5	79.6	42.2			0.254	0.237
0.10	81.7	75.9	21.7			0.201	0.190
0.15	71.2	68.3	5.84			0.150	0.144
0.20	64.2	62.6	2.41			0.129	0.126
0.30	55.2	54.4	0.72			0.109	0.107
0.40	49.2	48.8	0.32			0.0962	0.0954
0.50	44.7	44.5	0.18			0.0872	0.0868
0.60	41.3	41.2	0.11			0.0804	0.0802
0.80	36.3	36.2	0.05			0.0706	0.0704
1.0	32.6	32.5	0.03			0.0634	0.0632
1.5		26.4		0.10			0.0515
2.0		22.5		0.38			0.0444
3.0		17.73		1.08	0.002		0.0346
4.0		14.78		1.79	0.007		0.0322
5.0		12.75		2.38	0.02		0.0294
6.0		11.27		2.91	0.03		0.0276
8.0		9.22		3.75	0.06		0.0253
10		7.85		4.50	0.09		0.0242
15		5.81		5.86	0.17		0.0230
20		4.66		6.86	0.23		0.0228
30		3.39		8.23	0.32		0.0232
40		2.69		9.22	0.40		0.0239
50		2.24		9.92	0.45		0.0245
60		1.931		10.5	0.50		0.0251
80		1.522		11.4	0.58		0.0262
100		1.263		12.0	0.65		0.0270

* Data in the first column are given by the sum of coherent scattering and of incoherent scattering from the Klein-Nishina formula corrected for binding effects. In the second column incoherent scattering is given by the Klein-Nishina formula for free electrons.

† Barns/molecule $\times 0.001942 = \text{cm}^2/\text{g}$.

Table 8-28. Air

(0.755 N, 0.232 O, 0.013 A by weight mass-absorption coefficient)

Photon energy, Mev	Total		Photon energy, Mev	Total	
	With coherent, cm ² /g	Without coherent, cm ² /g		With coherent, cm ² /g	Without coherent, cm ² /g
0.01	5.09	4.89	1.0	0.0635	0.0635
0.015	1.59	1.48	1.5		0.0517
0.02	0.764	0.697	2.0		0.0445
0.03	0.349	0.317	3.0		0.0357
0.04	0.245	0.226	4.0		0.0307
0.05	0.204	0.194	5.0		0.0274
0.06	0.186	0.178	6.0		0.0250
0.08	0.166	0.161	8.0		0.0220
0.10	0.155	0.151	10		0.0202
0.15	0.136	0.134	15		0.0178
0.20	0.123	0.123	20		0.0166
0.30	0.107	0.106	30		0.0158
0.40	0.0954	0.0953	40		0.0156
0.50	0.0868	0.0868	50		0.0157
0.60	0.0804	0.0804	60		0.0158
0.80	0.0706	0.0706	80		0.0160
			100		0.0164

Table 8-29. Concrete *

(0.56% H, 49.56% O, 31.35% Si, 4.56% Al, 8.26% Ca, 1.22% Fe, 0.24% Mg, 1.71% Na, 1.92% K, 0.12% S) ($\rho = 2.35 \text{ g/cm}^3$)

Photon energy, Mev	Mass-absorption coefficient, cm ² /g	Photon energy, Mev	Mass-absorption coefficient, cm ² /g	Photon energy, Mev	Mass-absorption coefficient, cm ² /g
0.01	24.6	0.30	0.107	6.0	0.0268
0.015	7.68	0.40	0.0954	8.0	0.0243
0.02	3.34	0.50	0.0870	10.0	0.0229
0.03	1.10	0.60	0.0804	15	0.0214
0.04	0.542	0.80	0.0706	20	0.0209
0.05	0.350	1.0	0.0635	30	0.0209
0.06	0.267	1.5	0.0517	40	0.0213
0.08	0.197	2.0	0.0445	50	0.0217
0.10	0.169	3.0	0.0363	60	0.0222
0.15	0.139	4.0	0.0317	80	0.0230
0.20	0.124	5.0	0.0287	100	0.0237

* Coherent scattering is not included in the calculations.

particular barrier thickness and barrier material. For most shielding problems, the attenuation more nearly approaches broad-beam conditions; so these are the data most often desired.

Build-up factors may be determined for attenuation of the number of photons, of the intensity of the photons, of the energy absorbed from the photons, and of the exposure dose of photons. Usually the build-up factor is obtained experimentally for exposure dose and is therefore the one most often quoted.

In practice, the location of the barrier is between a radiation source and a position where a lower exposure dose is desired. In these cases, the source is often surrounded with a source shield to limit the radiation as far as practicable to the useful beam which emerges through an aperture in the source shield. Part of this useful beam may be scattered from the barrier or from other material in its path. Thus, barriers are required for the useful beam (primary protective barrier) and from the secondary radiation leaking through the source shield as well as from scattered radiation (secondary protective barriers). Generally, the source shield is thick enough so that the effective absorption coefficient is constant for additional attenuation.

There is a large amount of **theoretical attenuation data** available. One large class of such data is obtained on the assumption of an **infinite homogeneous medium** for simple sources. The earliest work in this class assumes that the Compton-scattering equation gives approximately the correct loss of energy, but the scattering photon is assumed either to be undeviated by the collision or to go off at the rms angle (Solon and Wilkins, Straight-ahead and Root Mean Square Angle Calculations for 20 Mc² Gamma-rays in Lead, *AEC Rept.* NYO-635, Dec. 15, 1950; Solon *et al.*, Gamma Transmissions in Iron, Tungsten, Lead, Uranium, and a Pure Compton Scatterer by Root-mean-square Angle Calculation, *AEC Rept.* NYO-637, Apr. 5, 1951). This method is more nearly correct for high-atomic-number materials where the large-angle scattered photons are absorbed photoelectrically and for high initial energies when the actual scattering is principally in the forward direction.

A second method of this class with an accuracy goal of 20 to 30 per cent considered the transmission of initially monoenergetic monodirectional gamma rays through iron and lead (Peebles, Gamma-ray Transmission through Finite Slabs, *J. Appl. Phys.*, vol. 24, p. 1272, 1953; vol. 24, p. 1437, 1953). This method classifies the photons into the number of scatterings they have experienced in a thin slab, and the probability of the transmission for each class is computed. The transmitted spectrum from this slab is then considered as the incident spectrum for the next slab. Data for barrier thicknesses of up to $\mu x = 20$ are available. The agreement with the moment method seems to be reasonably good.

A third method of this class has been developed by Spencer and Fano (Penetration and Diffusion of X-rays, *Phys. Rev.*, vol. 81, p. 464, 1951; *J. Research NBS*, vol. 46, p. 446, 1951). In this method, high accuracy is aimed for at appreciable distances from the source ($\mu x = 1$ to 20). This method consists of solving the equation of continuity for the radiation by the method of moments. Both the spectra and the build-up factors are determined by this method, but only the exposure dose build-up factors B_x are listed in Tables 8-30 through 8-40. (Goldstein and Wilkins, Calculations of the Penetration of Gamma-rays, *AEC*

Table 8-30. Dose Build-up Factor B_r

Point isotropic source

Water

E_0 , Mev	$\mu_0 r$						
	1	2	4	7	10	15	20
0.255	3.09	7.14	23.0	72.9	166	456	982
0.5	2.52	5.14	14.3	38.8	77.6	178	334
1.0	2.13	3.71	7.68	16.2	27.1	50.4	82.2
2.0	1.83	2.77	4.88	8.46	12.4	19.5	27.7
3.0	1.69	2.42	3.91	6.23	8.63	12.8	17.0
4.0	1.58	2.17	3.34	5.13	6.94	9.97	12.9
6.0	1.46	1.91	2.76	3.99	5.18	7.09	8.85
8.0	1.38	1.74	2.40	3.34	4.25	5.66	6.95
10.0	1.33	1.63	2.19	2.97	3.72	4.90	5.98

Table 8-31. Dose Build-up Factor B_r

Point isotropic source

Aluminum

E_0 , Mev	$\mu_0 r$						
	1	2	4	7	10	15	20
0.5	2.37	4.24	9.47	21.5	38.9	80.8	141
1.0	2.02	3.31	6.57	13.1	21.2	37.9	58.5
2.0	1.75	2.61	4.62	8.05	11.9	18.7	26.3
3.0	1.64	2.32	3.78	6.14	8.65	13.0	17.7
4.0	1.53	2.08	3.22	5.01	6.88	10.1	13.4
6.0	1.42	1.85	2.70	4.06	5.49	7.97	10.4
8.0	1.34	1.68	2.37	3.45	4.58	6.56	8.52
10.0	1.28	1.55	2.12	3.01	3.96	5.63	7.32

Table 8-32. Dose Build-up Factor B_r

Point isotropic source

Iron

E_0 , Mev	$\mu_0 r$						
	1	2	4	7	10	15	20
0.5	1.98	3.09	5.98	11.7	19.2	35.4	55.6
1.0	1.87	2.89	5.39	10.2	16.2	28.3	42.7
2.0	1.76	2.43	4.13	7.25	10.9	17.6	25.1
3.0	1.55	2.15	3.51	5.85	8.51	13.5	19.1
4.0	1.45	1.94	3.03	4.91	7.11	11.2	16.0
6.0	1.34	1.72	2.58	4.14	6.02	9.89	14.7
8.0	1.27	1.56	2.23	3.49	5.07	8.50	13.0
10.0	1.20	1.42	1.95	2.99	4.35	7.54	12.4

RADIATION ATTENUATION DATA

Table 8-33. Dose Build-up Factor B_r

Point isotropic source

Tin

E_0 , Mev	$\mu_0 r$						
	1	2	4	7	10	15	20
0.5	1.56	2.08	3.09	4.57	6.04	8.64	
1.0	1.64	2.30	3.74	6.17	8.85	13.7	18.8
2.0	1.57	2.17	3.53	5.87	8.53	13.6	19.3
3.0	1.46	1.96	3.13	5.28	7.91	13.3	20.1
4.0	1.38	1.81	2.82	4.82	7.41	13.2	21.2
6.0	1.26	1.57	2.37	4.17	6.94	14.8	29.1
8.0	1.19	1.42	2.05	3.57	6.19	15.1	34.0
10.0	1.14	1.31	1.79	2.99	5.21	12.5	33.4

Table 8-34. Dose Build-up Factor B_r

Point isotropic source

Tungsten

E_0 , Mev	$\mu_0 r$						
	1	2	4	7	10	15	20
0.5	1.28	1.50	1.84	2.24	2.61	3.12	
1.0	1.44	1.83	2.57	3.62	4.64	6.25	(7.35)
2.0	1.42	1.85	2.72	4.09	5.27	8.07	(10.6)
3.0	1.36	1.74	2.59	4.00	5.92	9.66	14.1
4.0	1.29	1.62	2.41	4.03	6.27	12.0	20.9
6.0	1.20	1.43	2.07	3.60	6.29	15.7	36.3
8.0	1.14	1.32	1.81	3.05	5.40	15.2	41.9
10.0	1.11	1.25	1.64	2.62	4.65	14.0	39.3

Table 8-35. Dose Build-up Factor B_r

Point isotropic source

Lead

E_0 , Mev	$\mu_0 r$						
	1	2	4	7	10	15	20
0.5	1.24	1.42	1.69	2.00	2.27	2.65	(2.73)
1.0	1.37	1.69	2.26	3.02	3.74	4.81	5.86
2.0	1.39	1.76	2.51	3.66	4.84	6.87	9.00
3.0	1.34	1.68	2.43	3.75	5.30	8.44	12.3
4.0	1.27	1.56	2.25	3.61	5.44	9.80	16.3
5.1097	1.21	1.46	2.08	3.44	5.55	11.7	23.6
6.0	1.18	1.40	1.97	3.34	5.69	13.8	32.7
8.0	1.14	1.30	1.74	2.89	5.07	14.1	44.6
10.0	1.11	1.23	1.58	2.52	4.34	12.5	39.2

Table 8-36. Dose Build-up Factor B_r

Plane monodirectional source

Water

E_0 , Mev	$\mu_0 x$					
	1	2	4	7	10	15
0.5	2.63	4.29	9.05	20.0	35.9	74.9
1.0	2.26	3.39	6.27	11.5	18.0	30.8
2.0	1.84	2.63	4.28	6.96	9.87	14.4
3.0	1.69	2.31	3.57	5.51	7.48	10.8
4.0	1.58	2.10	3.12	4.63	6.19	8.54
6.0	1.45	1.86	2.63	3.76	4.86	6.78
8.0	1.36	1.69	2.30	3.16	4.00	5.47

Table 8-37. Dose Build-up Factor B_r

Plane monodirectional source

Iron

E_0 , Mev	$\mu_0 x$					
	1	2	4	7	10	15
0.5	2.07	2.94	4.87	8.31	12.4	20.6
1.0	1.92	2.74	4.57	7.81	11.6	18.9
2.0	1.69	2.35	3.76	6.11	8.78	13.7
3.0	1.58	2.13	3.32	5.26	7.41	11.4
4.0	1.48	1.90	2.95	4.61	6.46	9.92
6.0	1.35	1.71	2.48	3.81	5.35	8.39
8.0	1.27	1.55	2.17	3.27	4.58	7.33
10.0	1.22	1.44	1.95	2.89	4.07	6.70

Table 8-38. Dose Build-up Factor B_r

Plane monodirectional source

Tin

E_0 , Mev	$\mu_0 x$					
	1	2	4	7	10	15
1.0	1.65	2.24	3.40	5.18	7.19	10.5
2.0	1.58	2.13	3.27	5.12	7.13	11.0
4.0	1.39	1.80	2.69	4.31	6.30	
6.0	1.27	1.57	2.27	3.72	5.77	11.0
10.0	1.16	1.33	1.77	2.81	4.53	9.68

Table 8-39. Dose Build-up Factor B_r Plane monodirectional source Lead

E_0 , Mev	$\mu_0 x$					
	1	2	4	7	10	15
0.5	1.24	1.39	1.63	1.87	2.08	
1.0	1.38	1.68	2.18	2.80	3.40	4.20
2.0	1.40	1.76	2.41	3.36	4.35	5.94
3.0	1.36	1.71	2.42	3.55	4.82	7.18
4.0	1.28	1.56	2.18	3.29	4.69	7.70
6.0	1.19	1.40	1.87	2.97	4.69	9.53
8.0	1.14	1.30	1.69	2.61	4.18	9.08
10.0	1.11	1.24	1.54	2.27	3.54	7.70

Table 8-40. Dose Build-up Factor B_r Plane monodirectional source Uranium

E_0 , Mev	$\mu_0 x$					
	1	2	4	7	10	15
0.5	1.17	1.28	1.45	1.60	1.73	
1.0	1.30	1.53	1.90	2.32	2.70	3.60
2.0	1.33	1.62	2.15	2.87	3.56	4.89
3.0	1.29	1.57	2.13	3.02	3.99	5.94
4.0	1.25	1.49	2.02	2.94	4.06	6.47
6.0	1.18	1.37	1.82	2.74	4.12	7.79
8.0	1.13	1.27	1.61	2.39	3.65	7.36
10.0	1.10	1.21	1.48	2.12	3.21	6.58

Rept. NYO-3075, Jan. 3, 1954.) Results from both a point source emitting isotropically and a plane source emitting perpendicular to the sources are indicated. The value of B_r decreases for higher Z because more of the scattered radiation is absorbed photoelectrically. This decrease is more important at the lower energies. Interpolation for other materials of atomic number Z and initial energy E_0 is possible, but linear interpolation is usually not adequate. If the spectrum of radiation is complex, then each component should be treated separately to determine the over-all attenuation. In case the barrier consists of a homogeneous mixture of different elements and no boundaries are present, it has been suggested that an effective value of Z be used. The absorption coefficient of this Z should approximate that of the mixture.

Sometimes a thin layer of high- Z material follows a principal barrier of low- Z material. In this case the low-energy scattered radiation from the low- Z material is photoelectrically absorbed by the high- Z material. Hence an approximate solu-

tion can be obtained by assuming that the build-up factor is that which would result from having the entire penetration in the high-Z material.

According to the authors, the tabulated values in Tables 8-30 to 8-40 are probably accurate to within ± 5 per cent on the average, but with the possibility of any particular value being in error by as much as ± 10 per cent, especially when the build-up factors are sensitive to low-energy spectra where the absorption coefficients are less well known. Interpolation to other values of E_0 or Z may double this error. An extension to other geometries of source or media may lead to errors of a factor of 2 in the build-up factor.

In order to simplify computations, the build-up factor sometimes can be expressed adequately as either a linear function of the distance or the sum of two exponential terms. (Taylor, Application of Gamma-ray Build-up Data to Shield Design, *WAPD Memo Rm 217*, Jan. 25, 1954.)

Another method of attenuation computation is the **Monte Carlo technique**. In this method, the histories of many photons are followed using known absorption coefficients and scattered photon energies for particular geometries. As the probability of penetration for most barriers is an unlikely occurrence, some method must be used to favor those successful penetrations in order to reduce the computational effort. This method has been most successful for moderate penetrations when the attenuation is equal to or less than 10^4 . Such computations have indicated the effect of boundaries. Therefore, they can be used to correct the theoretical data on infinite barriers for boundaries.

Most of the experimental data are for conditions where the source and detector are outside the barrier. These results can be divided into those for the useful beam and those for the scattered radiation.

X-rays attenuation curves for the useful beam are generally given with an ordinate in terms of roentgens per milliamperemminute at 1 m in order to facilitate computations. Small amounts of radiation filtration used in most applications do not materially affect the barrier thickness required. Figures 8-3 to 8-9 show such curves for broad-beam conditions with concrete (density = 2.35 g/cm^3 or 147 lb/cu ft) and for lead barriers.

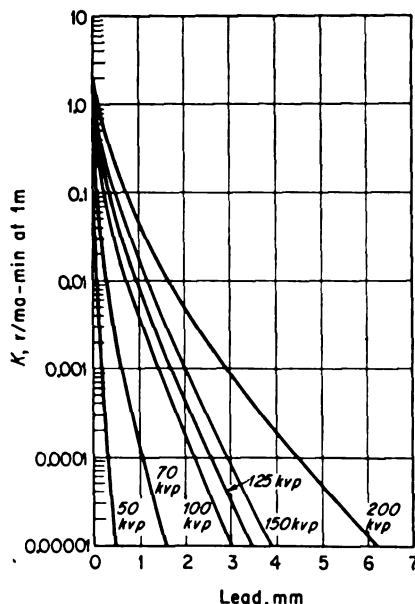


FIG. 8-3. Attenuation in lead of X-rays produced by potentials of 50 to 200 kv peak.

The measurements were made with a 90° angle between the electron beam and the axis of the X-ray beam and with a pulsed waveform. The curves at 50 and 70 kv peak were obtained by interpolation and extrapolation of available data (Braestrup, *Radiology*, vol. 43, p. 286, 1944). The filtrations were 0.5 mm of aluminum for 50, 70, 100, and 125 kv peak, and 3 mm of aluminum for 150 and 200 kv peak. (From *NBS Handbook 60*, U.S. Government Printing Office, Washington, D.C., 1955.)

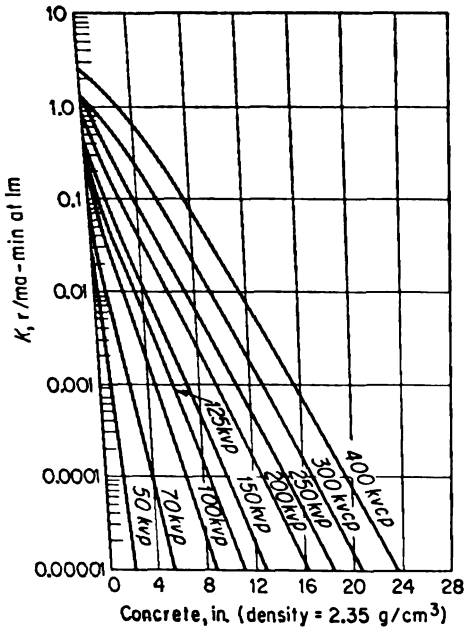


FIG. 8-4. Attenuation in concrete of X-rays produced by potentials of 50 to 400 kv.

The measurements were made with a 90° angle between the electron beam and the axis of the X-ray beam. The curves for 50 to 250 kv peak are for a pulsed waveform. The filtrations were 1 mm of aluminum for 50 kv peak, $1\frac{1}{2}$ mm of aluminum for 70 kv peak, 2 mm of aluminum for 100 kv peak, and 3 mm of aluminum for 125 to 250 kv peak [Trout *et al.*, *Radiology*, vol. 64, p. 114 (A), 1955, and private communication]. The 300- and 400-kvcp curves were interpolated from data obtained with a constant-potential generator and inherent filtration of approximately 3 mm of copper (Miller and Kennedy, *Radiology*, vol. 65, p. 920, 1955). (From NBS Handbook 60, U.S. Government Printing Office, Washington, D.C., 1955.)

FIG. 8-5. Attenuation in lead of X-rays produced by potentials of 250 to 400 kv.

The measurements were made with a 90° angle between the electron beam and the axis of the X-ray beam. The 250-kv-peak curve is for a pulsed waveform and a filtration of 3 mm of aluminum (Braestrup, *Radiology*, vol. 43, p. 286, 1944). The 300- and 400-kvcp curves were interpolated from data obtained with a constant-potential generator and inherent filtration of approximately 3 mm of copper (Miller and Kennedy, *Radiology*, vol. 65, p. 920, 1955). (From NBS Handbook 60, U.S. Government Printing Office, Washington, D.C., 1955.)

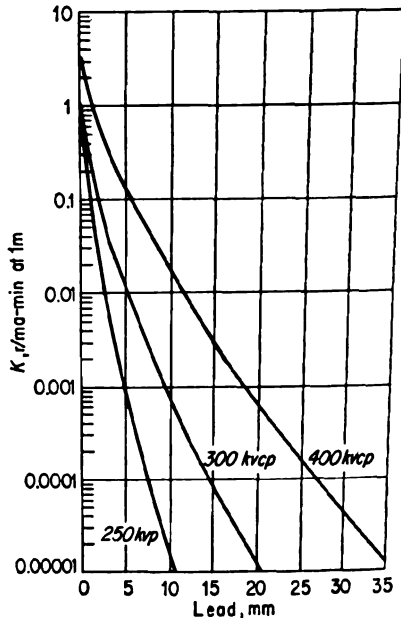


Fig. 8-6. Attenuation in lead of X-rays produced by potentials of 500 to 2,000 kvcp.

The measurements were made with a 0° angle between the electron beam and the axis of the X-ray beam and with a constant-potential generator. The 500- and 1,000-kvcp curves were obtained with filtration of 2.8 mm of tungsten, 2.8 mm of copper, 2.1 mm of brass, and 18.7 mm of water (Wyckoff *et al.*, *Radiology*, vol. 51, p. 849, 1948). The 2,000-kvcp curve was obtained by extrapolating to broad-beam conditions (E. E. Smith) the data of Evans *et al.* (*Radiology*, vol. 85, p. 560, 1952; curves from this paper were extrapolated to broad-beam conditions by E. E. Smith, National Physical Laboratory, England). The inherent filtration was equivalent to 6.8 mm of lead. (From NBS Handbook 60, U.S. Government Printing Office, Washington, D.C., 1955.)

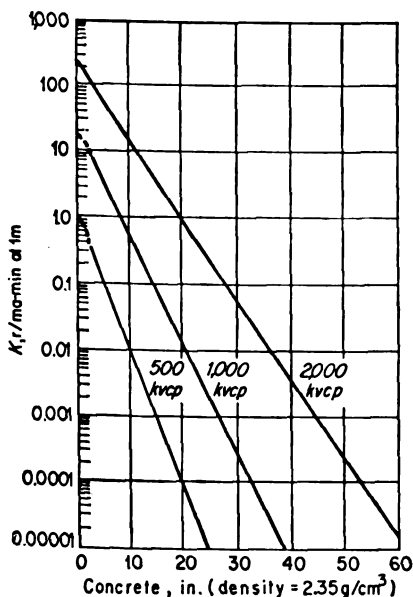
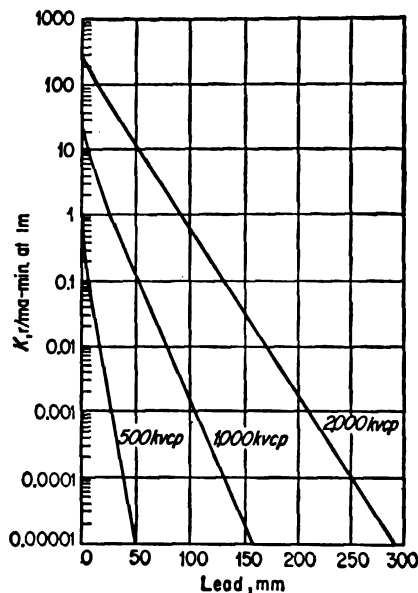


Fig. 8-7. Attenuation in concrete of X-rays produced by potentials of 500 to 2,000 kvcp.

The measurements were made with a 0° angle between the electron beam and the axis of the X-ray beam and with a constant-potential generator. The 500- and 1,000-kvcp curves were obtained with filtration of 2.8 mm of tungsten, 2.8 mm of copper, 2.1 mm of brass, and 18.7 mm of water (Wyckoff *et al.*, *Radiology*, vol. 51, p. 849, 1948). The 2,000-kvcp curve was obtained by extrapolating to broad-beam conditions (E. E. Smith) the data of Evans *et al.* (*Radiology*, vol. 85, p. 560, 1952). The inherent filtration was equivalent to 6.8 mm of lead. (From NBS Handbook 60, U.S. Government Printing Office, Washington, D.C., 1955.)

crete of X-
of 50 to 400

ade with a
beam and
The curves
used were
1 mm of
2 mm of
n of alumi-
n of alumi-
out *et al.*,
1955, and
300- and
lated from
t-potential
ion of ap-
Miller and
(20, 1955).
Government
, 1955.)

30 35

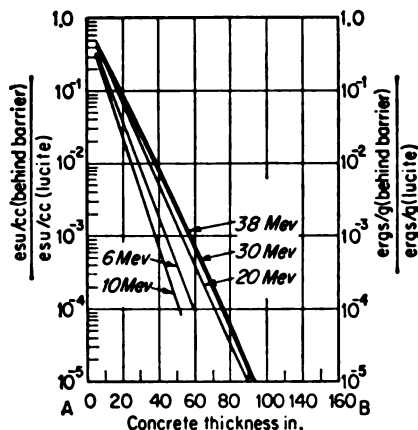


FIG. 8-8. Attenuation in concrete at betatron energies of 6, 10, 20, 30, 38 Mev.

Density of concrete, 147 lb/ft³. Scale A: The esu/cm³ detected in an ionization chamber behind various thicknesses of concrete, per esu/cm³ detected in a thimble chamber embedded in an 11.5-cm lucite cube at the same position without barrier, is plotted for various betatron energies. Scale B: The ratio of maximum tissue dose received with and without a concrete barrier calculated from scale A assuming the same conversion of 93 ergs/g for 1 esu/cm³. (From NBS Handbook 55, U.S. Government Printing Office, Washington, D.C., 1954. These data were taken from results of Kirn, Kennedy, and Wyckoff, *Radiology*, vol. 63, p. 94, 1954.)

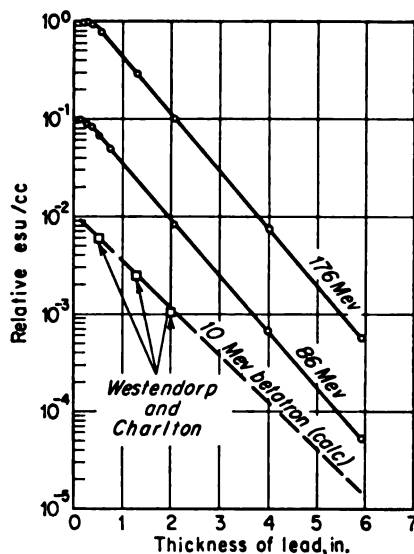


FIG. 8-9. Attenuation curves in lead.

The dashed curve has been calculated using a 10-Mev betatron spectrum. The amplitudes of the curves have been normalized so that the peak values coincide with the decade lines. (From Miller and Kennedy, *Radiation Research*, vol. 4, p. 380, 1956.)

There are also a large class of **low-atomic-number shielding materials** in which the absorption is principally by the Compton effect. For most of these no attenuation data are available, but it may be assumed that the amount of the attenuation is approximately proportional to the density of the material. Table 8-41 lists such

Table 8-41. Shielding Materials

Material	Density range, g/cm ³	Density of average sample, g/cm ³	Remarks
Lead:			
Lead sheet.....		11.35	Used in cinder blocks or glued to plywood or lath
Lead glass.....	3-6		For observation windows or for protective gowns
Lead rubber.....	3-5	5	For garments and other uses where flexibility is required
Masonry:			For structural shielding *
Brick.....	1.6-2.5	1.9	
Barium concrete.....	2.7-3.2		
Granite.....	2.6-2.7	2.63	
Limestone.....	1.87-2.69	2.30	
Marble.....	2.47-2.86	2.70	
Plaster.....		1.54	
Sandstone.....	1.90-2.69	2.20	
Siliceous concrete.....	2.25-2.40	2.35	
Tile.....	1.6-2.5	1.9	
Steel.....		7.9	For structural and portable shielding
Plate glass.....		2.5	For structural shielding

* Obtain thickness requirement for all except barium concrete by multiplying the calculated concrete thickness by $2.35/\rho$ where ρ is density (in g/cm³) of masonry.

materials, their range of densities, and the average density for each (Braestrup and Wyckoff, "Radiation Protection," Charles C Thomas, 1958). To use these data, the thickness of a concrete barrier is first computed, and this thickness is multiplied by $2.35/d$ where d is the density in g/cm³ of the desired material. This scheme is adequate for materials of as high an atomic number as iron when using Cesium-137 radiation, but the photoelectric effect produces an appreciable error with iron for Gold-198 radiation. For marble (which is really CaCO₃) the attenuation on a density basis is not significantly different even for 100-kv X-rays.

Figures 8-10 to 8-12 show the transmission of **normally incident radiation** (from radioactive cesium, cobalt, and radium gamma) for concrete, iron, and lead barriers under broad-beam conditions (Protection against Radiations from Radium, Cobalt-60, and Cesium-137, National Bureau of Standards Handbook 54, GPO, Washing-

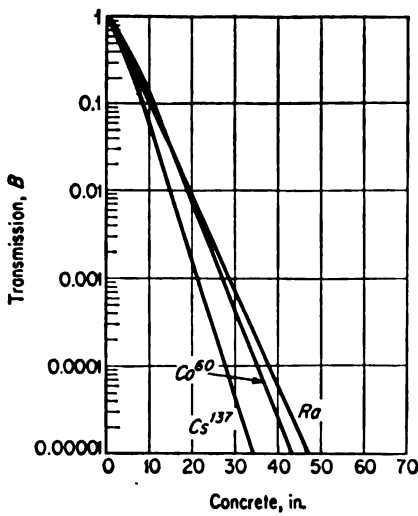


FIG. 8-10. Transmission through concrete (specific gravity 147 lb/ft³) of gamma rays from radium, Cobalt-60, and Cesium-137. (From NBS Handbook 54, U.S. Government Printing Office, Washington, D.C., 1954.)

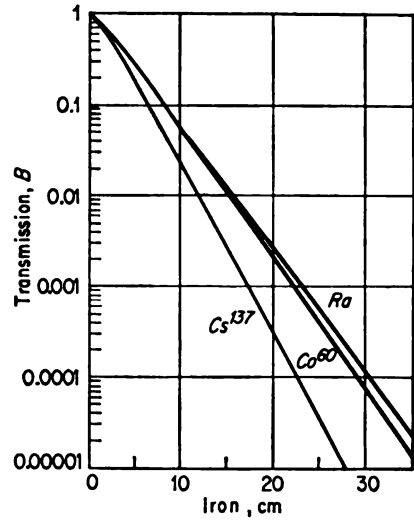


FIG. 8-11. Transmission through iron of gamma rays from radium, Cobalt-60, and Cesium-137. (From NBS Handbook 54, U.S. Government Printing Office, Washington, D.C., 1954.)

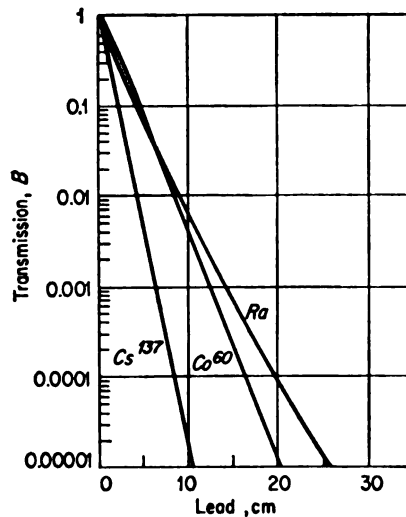


FIG. 8-12. Transmission through lead of gamma rays from radium, Cobalt-60, and Cesium-137. (From NBS Handbook 54, U.S. Government Printing Office, Washington, D.C., 1954.)

ton, 1954). Figures 8-13 to 8-18 show the effect of **obliquely incident radiation** on the transmission through concrete and lead barriers (Kirn and Kennedy, *The Attenuation of Gamma Rays at Oblique Incidence*, *Radiology*, vol. 63, p. 94, 1954). Here the oblique thickness is the normal thickness divided by the cosine of the angle the ray makes with the normal.

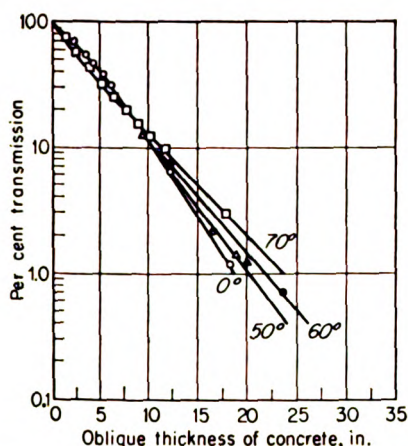


FIG. 8-13. Transmission of Cobalt-60 radiation through concrete of density 147 lb/ft³ for indicated angles of incidence. (From Kirn, Kennedy, and Wyckoff, *Radiology*, vol. 63, p. 94, 1954.)

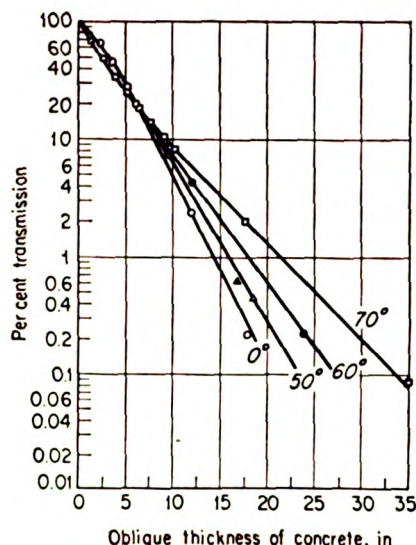


FIG. 8-14. Transmission of Cesium-137 radiation through concrete of density 147 lb/ft³ for indicated angles of incidence. (From Kirn, Kennedy, and Wyckoff, *Radiology*, vol. 63, p. 94, 1954.)

Shielding computations for **primary protective barriers** take into account the combined effects of (1) the distance d in feet between the source and the position to be shielded, (2) whether the position is inside or outside of a controlled area, (3) the interposed-barrier transmission K_p , (4) the **use factor** U (the fraction of the time the beam may be directed toward a given barrier), (5) the fractional **occupancy factor** T , and (6) the weekly **work load** W . The design level for controlled areas is 0.1 rem/week and for outside controlled areas is 0.01 rem/week (see Sec. 3). Table 8-42 lists some values of U for selected applications. Table 8-43 gives values

Table 8-42. Suggested Use Factors U for Design Purposes

Purpose	Ceiling	Walls *	Floor
Medical radiography...	$\frac{1}{6}$	$\frac{1}{4}$	1
Dentistry.....	$\frac{1}{6}$	$\frac{1}{4}$	$\frac{1}{6}$
Therapy.....	$\frac{1}{6}$	$\frac{1}{4}$	1

* Special conditions such as rotating-chair therapy may modify these values.

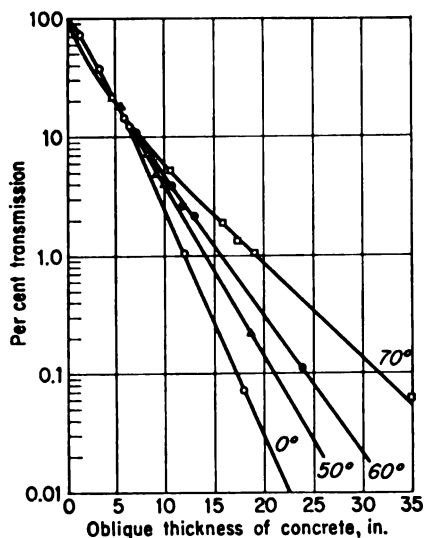


FIG. 8-15. Transmission of Gold-198 radiation through concrete of density 147 lb/ft³ for indicated angles of incidence. (From Kirn, Kennedy, and Wyckoff, *Radiology*, vol. 63, p. 94, 1954.)

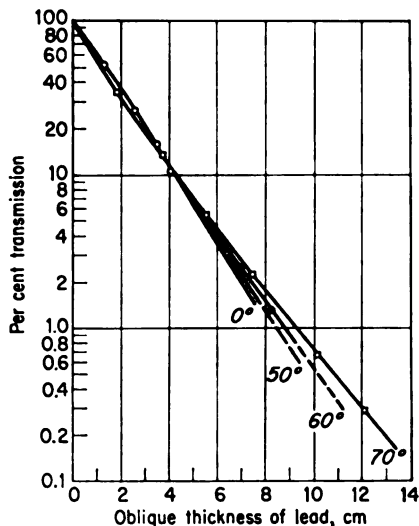


FIG. 8-16. Transmission of Cobalt-60 radiation through lead for the indicated angles of incidence. (From Kirn, Kennedy, and Wyckoff, *Radiology*, vol. 63, p. 94, 1954.)

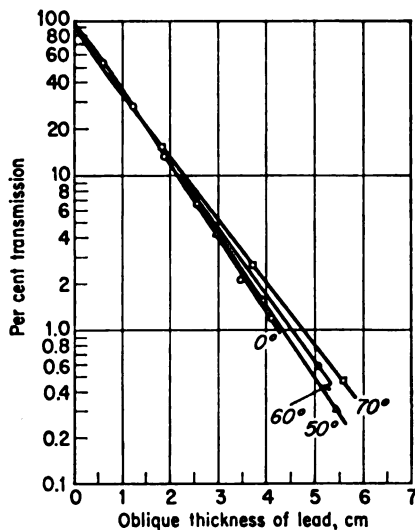
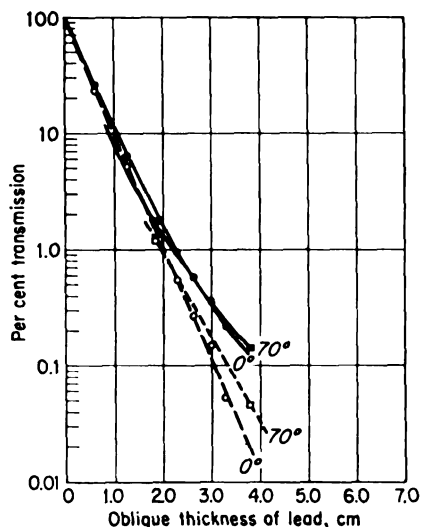


FIG. 8-17. Transmission of Cesium-137 radiation through lead for the indicated angles of incidence. (From Kirn, Kennedy, and Wyckoff, *Radiology*, vol. 63, p. 94, 1954.)

FIG. 8-18. Transmission of Gold-198 radiation through lead for 0° and 70° angles of incidence.

The dashed curves have been corrected for the presence of the 1.09 Mev and 0.67 Mev gold lines and hence are the transmission of the 0.411 Mev line only. Solid curves are the uncorrected data. (From Kirn, Kennedy, and Wyckoff, *Radiology*, vol. 63, p. 94, 1954.)



of T (Braestrup and Wyckoff, "Radiation Protection," Charles C Thomas, 1958). For X-ray sources of less than 2 million volts, W is usually expressed in terms of

Table 8-43. Occupancy Factors

(For use as a guide in planning shielding where complete occupancy data are not available)

Full occupancy ($T = 1$):

Control space, wards, office workrooms, darkrooms, corridors and waiting space large enough to hold desks, rest rooms used by the radiologic staff and others routinely exposed to radiation, play areas, occupied rooms in adjacent buildings

Partial occupancy ($T = \frac{1}{4}$):

Rest rooms not used by radiologic personnel, unattended parking lots, utility rooms

Occasional occupancy ($T = \frac{1}{16}$):

Stairways, automatic elevators, streets, closets too small for future workrooms, toilets not used by radiologic personnel

ma-min/week, for multimegavolt X-ray installations in esu/cm³ (lucite)/week at 1 m, and for gamma-ray sources in r/week at 1 m.

Thus, for X-rays of 2 million volts or less the transmission K in ma-min at 1 m is

$$K = \frac{0.01d^2}{WUT} \quad (8-3)$$

for controlled areas and

$$K = \frac{0.001d^2}{WUT} \quad (8-4)$$

for outside controlled areas.

Values of K are obtained by substituting in Eq. (8-3) or (8-4) the pertinent values of d , W , U , and T . Figures 8-3 through 8-7 may be used to find the barrier thickness corresponding to the computed value for K .

For X-ray sources of greater than 2 million volts and for γ -ray sources (e.g., radium and Cobalt-60), the same equations hold but K is changed to B where B is the fractional transmission. Figures 8-8 through 8-12 assist in determining barrier requirements for these cases.

Secondary protective barriers shield against the leakage and scattered radiation. As these two radiations are generally of different quality, the shielding required to reduce each to the design level is usually computed separately. If the resulting computed barrier thicknesses are nearly equal, one half-value layer (HVL) is added to the greater thickness to obtain the required barrier thickness; if one is 3 HVL or more larger than the other, it alone is the required thickness.

For **leakage radiation** the permissible transmission T_L of the barrier is the ratio of the design level P to the leakage radiation at a distance d (feet)

$$T_L = \frac{Pd^2}{L(3.28)^2} \quad (8-5)$$

where L is the leakage radiation at 1 m from the source. The permissible leakage has been agreed upon for diagnostic X-ray source shields (0.1 r/hr at 1 m for maximum continuous rated current at maximum kilovolts) for therapeutic X-ray tube shields of 2 million volts and below (1.0 r/hr at 1 m for maximum continuous rated current at maximum kilovolts) and for teletherapy units (0.001W). The permissible leakage radiation has not been standardized above 2 million volts; so this information must be obtained from the manufacturer.

Because the leakage radiation for diagnostic units is generally negligible compared with the scattered radiation, it is not necessary to consider it. For the other two applications,

$$(T_L)_X = \frac{2d^2I}{3WT} \quad (8-6)$$

$$(T_L)_\gamma = \frac{10d^2}{WT} \quad (8-7)$$

for controlled areas, where I is the maximum continuous rated current in milliamperes at the maximum X-ray tube voltage, and

$$(T_L)_X = \frac{d^2I}{15WT} \quad (8-8)$$

$$(T_L)_\gamma = \frac{d^2}{WT} \quad (8-9)$$

for outside controlled areas.

As the source shield has generally hardened the radiation so that it has a constant HVL, it is convenient to calculate T_L for Eqs. (8-5) to (8-9) and find the corresponding number of half-value layers from Fig. 8-19. The product of the number of

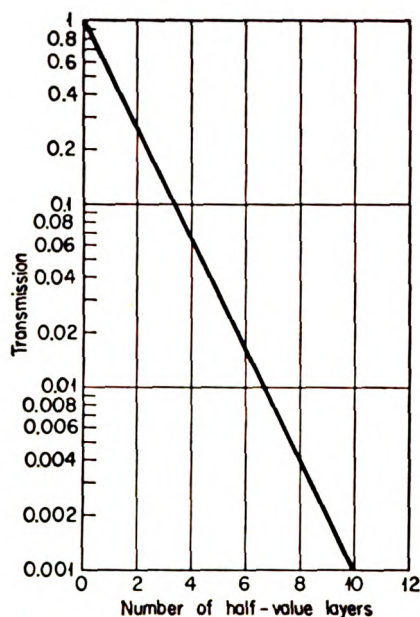


Fig. 8-19. Relation between transmission and number of half-value layers.

half-value layers and the HVL for the barrier material and radiation quality (from Table 8-44) gives the barrier thickness for leakage radiation.

Table 8-44. Half-value Layer *

Attenuating material	HVL for a tube potential of										
	50 kvp	70 kvp	100 kvp	125 kvp	150 kvp	200 kvp	250 kvp	300 kvep	400 kvep	1,000 kvep	2,000 kvep
Lead, mm. . . .	0.05	0.18	0.24	0.27	0.3	0.5	0.8	1.5	2.2	8.0	12
Concrete, in.	0.2	0.5	0.7	0.8	0.9	1.0	1.1	1.2	1.3	1.8	2.45

* Approximate values determined at high filtration.

The principal source of **scattered radiation** is the first object struck by the useful beam. The amount of scattered radiation depends upon the irradiated area, the angle of scattering, the incident exposure dose, and the amount of scattered radiation absorbed in the object. Table 8-45 lists some results quoted in the literature for different beam sizes, different sources, and a scattering angle of 90°. The scattering is slightly smaller for larger angles but increases for smaller angles especially at the higher energies. However, small-angle scattering often penetrates a barrier obliquely and may even strike a primary barrier. From the table, it is seen that the dose rate of the scattered radiation measured at 1 m from the scatterer

usually is not more than 0.03 per cent of that incident on the scatterer for each 100 cm² of field. Thus at a distance d (ft) from the scatterer, the radiation dose rate usually is not more than $0.003A/d^2$ of that incident on the scatterer, where A is the field in units of 100 cm².

Table 8-45. Radiation Scattered at 90° to the Useful Beam

Source	Literature reference	Field size	Per cent * of incident beam
75 kvp X-ray	(a)	25 × 18 cm	0.1
80 kvp X-ray	(b)	8 cm diam	0.002
80 kvp X-ray	(c)	10 × 15 cm	0.05
100 kvp X-ray	(c)	10 × 15 cm	0.05
200 kvp X-ray	(a)	15 cm diam	0.04
200 kvp X-ray	(d)	6 × 8 cm	0.034
1 Mvp X-ray	(e)	20 × 20 cm	0.076
2 Mvp X-ray	(f)	20 × 20 cm	0.07
Co ⁶⁰	(g)	26 × 26 cm	0.1

* Measured at 1 m from scatterer.

^a Behnken, H. C., Dosemetrische Untersuchungen über Röntgenstrahlenschutz und Strahlenschutzrohren, *Fortschr. Gebiete Röntgenstrahlen*, vol. 41, p. 245, 1931.

^b Braestrup, C. B., X-ray Protection in Diagnostic Radiology, *Radiology*, vol. 38, p. 207, 1942.

^c Keane, B. E., and G. Speigler, Stray Radiation from Diagnostic Beams, *Brit. J. Radiol.*, vol. 24, p. 198, 1951.

^d Braestrup, C. B., A Stray Radiation Survey of Twenty High-voltage Roentgen Installations, *Radiology*, vol. 31, p. 206, 1938.

^e Braestrup, C. B., and H. O. Wyckoff, Protection Requirements of One Million-volt (1 Mv) and Two Million-volt (2 Mv) Roentgen Ray Installations, *Radiology*, vol. 51, p. 840, 1948.

^f Wilson, C. W., and B. J. Perry, Physical Observations Relating to the 2 Mev Van de Graaf Electrostatic Generator at Westminster Hospital, *Brit. J. Radiol.*, vol. 25, p. 210, 1952.

^g Dixon, W. R., C. Garrett, and A. Morrison, Room-protection Measurements for Cobalt-60 Teletherapy Units, *Nucleonics*, vol. 10, no. 3, p. 42, 1952.

Most of the scattered radiation arises from the Compton effect. For low energies, the scattered radiation is not much smaller than the incident radiation (see Sec. 7), but for very high energies it attains a maximum energy of about 0.5 Mev for 90° scattering. Few data are available on the attenuation of scattered X-radiation. For X-rays of up to 500 kv, it is usually assumed (NBS Handbook 60) that the attenuation is the same as the useful beam and, for 500 to 2,000 kv, is similar to that for 500-kv radiation.

For a field size of about 400 cm² at 1 m from the source and for scattered X-radiation of 500 kv and below, the barrier transmission

$$K = \frac{10d^2}{TW} \quad (8-10)$$

for controlled areas, and

$$K = \frac{d^2}{TW} \quad (8-11)$$

for outside controlled areas.

For 1,000-kv X-rays (using 500-kv transmission curve)

$$K = \frac{0.5d^2}{TW} \quad (8-12)$$

for controlled areas, and

$$K = \frac{0.05d^2}{TW} \quad (8-13)$$

for outside controlled areas.

For 2,000-kv X-rays (using 500-kv transmission curves)

$$K = \frac{0.1d^2}{TW} \quad (8-14)$$

for controlled areas, and

$$K = \frac{0.01d^2}{TW} \quad (8-15)$$

for outside controlled areas.

The permissible barrier transmission is computed from Eqs. (8-10) to (8-15), and the corresponding barrier thickness obtained from Figs. 8-3 to 8-6.

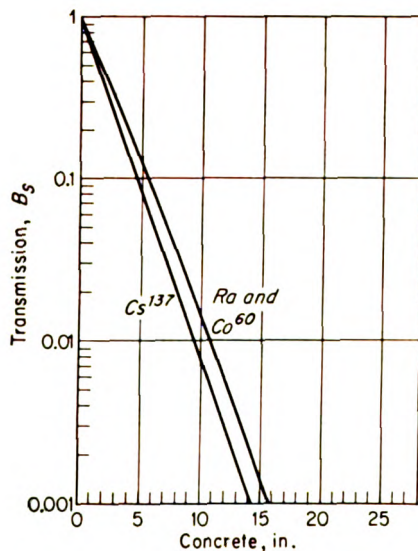


Fig. 8-20. Transmission through concrete (density 147 lb/ft³) of 90° scattered gamma rays from radium, Cobalt-60, and Cesium-137. (From NBS Handbook 54, U.S. Government Printing Office, Washington, D.C., 1954.)

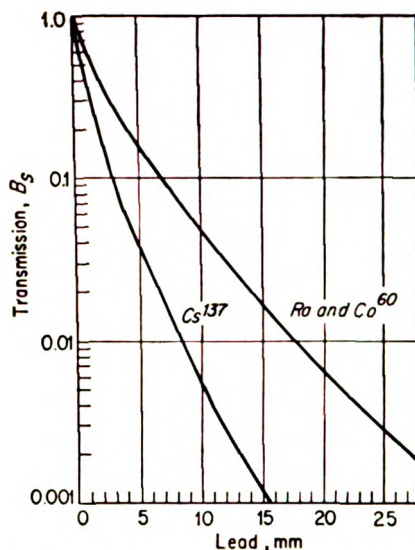


FIG. 8-21. Transmission through lead of 90° scattered gamma rays from radium, Cobalt-60, and Cesium-137. (From NBS Handbook 54, U.S. Government Printing Office, Washington, D.C., 1954.)

In gamma-ray sources, the barrier thickness (Figs. 8-20 and 8-21) required for 90° scattered radiation and a field size of 400 cm² at 1 m is given by the transmission

$$B_s = \frac{10d^2}{TW} \quad (8-16)$$

for controlled areas, and

$$B_s = \frac{d^2}{TW} \quad (8-17)$$

for outside controlled areas.

In a few cases (Figs. 8-22 through 8-25), more detailed information is available. These data were obtained for a field size of about 400 cm² at 1 m, where D_s/D_u is the ratio of scattered to incident dose rate. Equation 8-16 then becomes

$$1,000B_s \frac{D_s}{D_u} = \frac{10d^2}{TW} \quad (8-18)$$

$$1,000B_s \frac{D_s}{D_u} = \frac{d^2}{TW} \quad (8-19)$$

As before, the right sides of these equations yield measured values for particular geometries, and the curves of Figs. 8-22 through 8-25 give the corresponding barrier requirements.

X-ray barrier requirements are in Tables 8-46 to 8-62 for certain assumed conditions (Braestrup and Wyckoff, "Radiation Protection," Charles C Thomas,

1958). Auxiliary information (Smith and Kennedy, Dose to Walls in Radiographic Rooms, *Radiology*, vol. 64, p. 114, 1955) indicates that the methods used for the computation of barrier requirements for diagnostic purposes are providing considerably excess shielding for controlled areas. Probably most of the difference between the computed and required barrier thicknesses is due to the use of lower

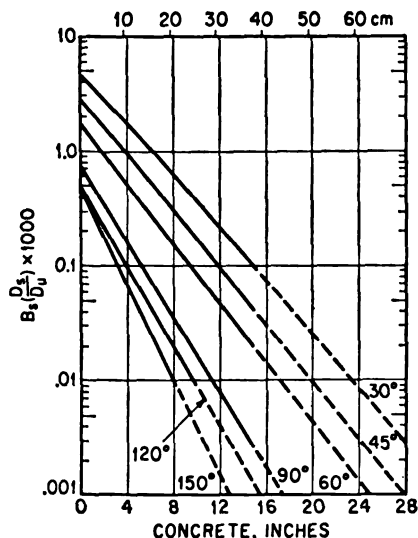


FIG. 8-22. Transmission through concrete of scattered Cobalt-60 radiation.

These data were obtained with a cylindrical masonite phantom and include radiation scattered from the phantom as well as from the diaphragm. The ordinate is the product of the fractional transmission in the barrier of the scattered radiation B_s , multiplied by the ratio of doses $\frac{D_s}{D_u}$ of the unattenuated scattered beam at 1 m to the useful beam at the scatterer. (From Mooney and Braestrup, U.S. Atomic Energy Commission Report NYO-2165, October, 1957.)

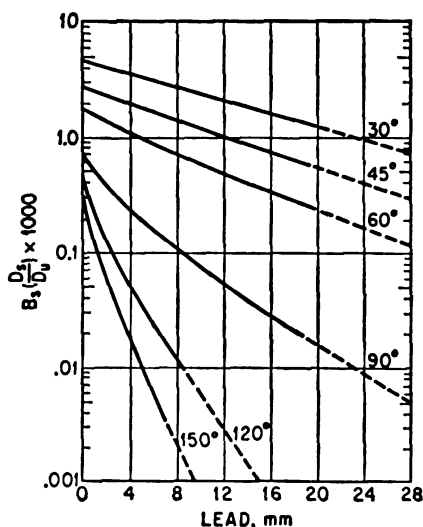


FIG. 8-23. Transmission through lead of scattered Cobalt-60 radiation.

These data were obtained with a cylindrical masonite phantom and include radiation scattered from the phantom as well as from the diaphragm. The ordinate is the product of the fractional transmission in the barrier of the scattered radiation B_s , multiplied by the ratio of doses $\frac{D_s}{D_u}$ of the unattenuated scattered beam at 1 m to the useful beam at the scatterer. (From Mooney and Braestrup, U.S. Atomic Energy Commission Report NYO-2165, October, 1957.)

tube voltages than those considered in the calculations. Unless one is certain that higher-voltage techniques will not be used in the future, the barriers indicated in Tables 8-46 and 8-47 should be used.

Unless there is information to assume otherwise, the work loads indicated for X-rays of 2,000 kv and below are conservative, and the barriers indicated will be adequate. The actual leakage radiation and output of machines above 2,000 kv have not been standardized, so it is not possible to make definite recommendations there for the secondary barrier or to do more than indicate sample results for the primary barrier.

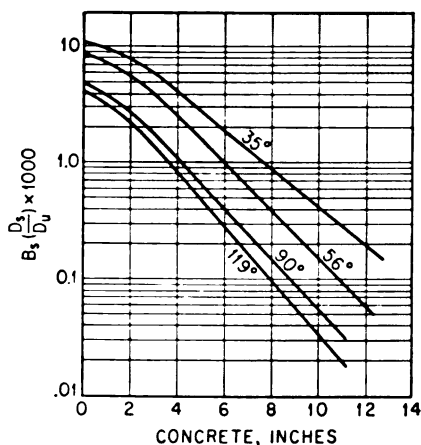


FIG. 8-24. Transmission through concrete of scattered Cesium-137 radiation.

The scatterer was an oblique concrete wall. The ordinate is the product of the fractional transmission in the barrier of the scattered radiation B_s , multiplied by the ratio of doses $\frac{D_s}{D_u}$ of the unattenuated scattered beam at 1 m to the useful beam at the scatterer. (Frantz and Wyckoff, *Radiology*, to be published.)

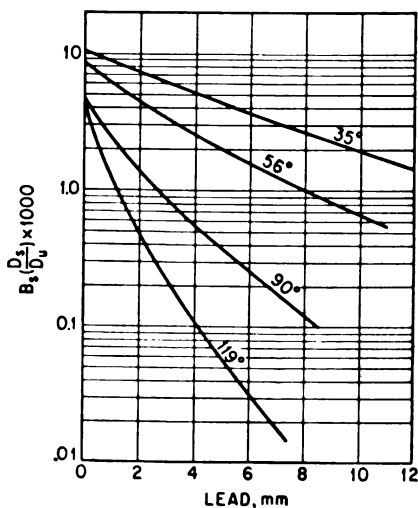


FIG. 8-25. Transmission through lead of scattered Cesium-137 radiation.

The scatterer was an oblique concrete wall. The ordinate is the product of the fractional transmission in the barrier of the scattered radiation B_s , multiplied by the ratio of doses $\frac{D_s}{D_u}$ of the unattenuated scattered beam at 1 m to the useful beam at the scatterer. (Frantz and Wyckoff, *Radiology*, to be published.)

Table 8-47. Shielding Requirements for Busy Radiographic Installations ^a

Type of barrier	U, T^b	Distance from tube to occupied area											
		5 ft (1.52 m)				7 ft (2.13 m)				10 ft (3.05 m)			
		Lead		Concrete		Lead		Concrete		Lead		Concrete	
		Mm	In. ^c	Psf ^d	In.	Mm	In. ^c	Psf ^d	In.	Mm	In. ^c	Psf ^d	In.
For controlled areas:													
	Primary	1	1.9	$\frac{1}{16}$	4.4	6.1	75	1.7	$\frac{1}{16}$	4.0	5.3	65	1.4
		$\frac{1}{2}$	1.4	$\frac{1}{16}$	3.3	4.6	56	1.2	$\frac{1}{32}$	2.8	3.8	47	1.0
		$\frac{1}{4}$	1.0	$\frac{1}{32}$	2.3	3.2	39	0.8	$\frac{1}{32}$	1.9	2.8	34	0.7
Secondary		1	0.5	...	1.2	1.7	21	0.4	...	0.9	1.4	17	0.2
										...	0.5	0.8	10
										...	...	...	0
										...	...	...	0
For environs:													
	Primary	1	2.7	$\frac{1}{32}$	6.3	8.1	99	2.4	$\frac{1}{32}$	6.6	7.2	88	2.2
		$\frac{1}{2}$	2.2	$\frac{1}{32}$	5.1	6.6	75	1.9	$\frac{1}{16}$	4.4	6.1	75	1.7
		$\frac{1}{4}$	1.7	$\frac{1}{16}$	4.0	5.3	65	1.5	$\frac{1}{16}$	3.5	5.0	61	1.3
Secondary		1	1.2	$\frac{1}{16}$	2.8	4.1	50	1.0	$\frac{1}{32}$	2.3	3.2	39	0.8
		$\frac{1}{2}$	0.8	$\frac{1}{32}$	1.9	2.8	34	0.6	$\frac{1}{32}$	1.4	2.0	25	0.4
										...	0.9	1.4	17
										...	...	...	0
Secondary		1	1.2	$\frac{1}{16}$	2.8	3.8	47	1.0	$\frac{1}{32}$	2.3	3.2	39	0.8
		$\frac{1}{2}$	0.8	$\frac{1}{32}$	1.9	2.8	34	0.6	$\frac{1}{32}$	1.4	2.0	25	0.4
										...	0.9	1.4	17
										...	...	...	0

^a For design levels of 100 mr/week for controlled areas and 10 mr/week for environs, and for a weekly work load of 1,000 ma-min at 100 kv, or 400 ma-min at 125 kv, or 200 ma-min at 150 kv.

^b U = use factor; T = occupancy factor. T is equal to 1 for controlled areas, and may be less than 1 for environs (see Table 8-43).

^c Approximately.

^d Pounds per square foot computed from millimeters of lead or inches of concrete and a concrete density of 2.35 g/cm³ (147 lb/ft³).

Table 8-48. Shielding Requirements for Busy Dental Installations ^a

Type of barrier	UT ^b	Distance from tube to occupied area											
		5 ft (1.52 m)				7 ft (2.13 m)				10 ft (3.05 m)			
		Lead		Concrete ^d		Lead		Concrete ^d		Lead		Concrete ^d	
		Mm	In. ^c	Mm	In. ^c	Mm	In. ^c	Mm	In. ^c	Mm	In. ^c	Mm	In. ^c
For controlled areas:													
Primary	1	0.8	1/32	2.8	1/32	0.7	1/32	2.4	1/32	0.6	1/32	2.1	1/32
	1/4	0.6	1/32	2.1	1/32	0.5	1/32	1.8	1/32	0.4	1/32	1.5	1/32
	1/16	0.4	...	1.5	...	0.3	...	1.1	...	0.2	...	0.8	...
Secondary	1	0.3	...	1.1	...	0.2	...	0.8	...	0	0	0	0
For environs:													
Primary	1	1.2	1/16	4.0	1/32	1.1	1/32	3.8	1/32	1.0	1/32	3.4	1/32
	1/4	1.0	1/32	3.4	1/32	0.9	1/32	3.1	1/32	0.8	1/32	2.8	1/32
	1/16	0.8	1/32	2.8	1/32	0.7	1/32	2.4	1/32	0.6	1/32	2.1	1/32
	1/64	0.6	1/32	2.1	1/32	0.5	1/32	1.8	1/32	0.4	...	1.5	1/32
	1/256	0.4	...	1.5	...	0.3	...	1.1	...	0.2	...	0.8	...
Secondary	1	0.6	1/32	2.1	1/32	0.5	1/32	1.8	1/32	0.4	...	1.5	1/32
	1/4	0.4	...	1.5	...	0.3	...	1.1	...	0.2	...	0.8	...
	1/16	0.2	...	0.8	...	0.1	...	0.4	...	0.1	...	0.4	0

^a For design levels of 100 mr/week for controlled areas and 10 mr/week for environs, and for a weekly work load of 800 ma-min at 70 kv.^b U = use factor; T = occupancy factor. T is equal to 1 for controlled areas, and may be less than 1 for environs (see Table 8-43).^c Approximately.^d Concrete density 2.35 g/cm³ (147 lb/ft³).

Table 8-49. Shielding Requirements for Busy 100-kv Therapeutic Installations ^a

Type of barrier	U ^b	Distance from tube to occupied area											
		5 ft (1.52 m)				7 ft (2.13 m)				10 ft (3.05 m)			
		Lead		Concrete		Lead		Concrete		Lead		Concrete	
		Mm	In. ^c	Psf ^d	In.	Mm	In. ^c	Psf ^d	In.	Mm	In. ^c	Psf ^d	In.
For controlled areas:	1	2.3	3/32	5.4	7.1	87	2.0	3/32	4.7	6.3	77	1.8	1/16
	Primary	1.8	1/16	4.2	5.7	70	1.6	1/16	3.7	5.2	64	1.4	1/16
	1/4	1.4	1/16	3.3	4.6	56	1.2	1/16	2.8	4.0	49	1.0	1/32
Secondary	1	1.3	1/16	3.0	4.3	53	1.1	1/32	2.6	3.7	45	0.9	1/32
	For environs:	3.2	1/8	7.5	9.2	113	2.9	1/8	6.8	8.5	104	2.7	1/8
	Primary	2.7	1/16	6.3	8.0	98	2.4	3/32	5.6	7.3	89	2.2	3/32
Secondary	1	2.2	3/32	5.1	6.8	83	1.9	1/16	4.4	6.0	74	1.7	1/16
	For environs:	1.7	1/16	4.0	5.5	67	1.4	1/16	3.3	4.6	56	1.2	1/16
	Primary	1.2	1/16	2.8	4.0	49	1.0	1/32	2.3	3.4	42	0.8	1/32
Secondary	1	2.1	3/32	4.9	6.6	81	1.8	1/16	4.2	5.7	70	1.6	1/16
	For environs:	1.6	1/16	3.7	5.2	64	1.4	1/16	3.3	4.6	56	1.2	1/16
	Primary	1.2	1/16	2.8	4.0	49	1.0	1/32	2.3	3.4	42	0.8	1/32

^a For design levels of 100 mr/week for controlled areas and 10 mr/week for environs, and for a weekly work load of 4,000 ma-min at 100 kvp.^b U = use factor. T = occupancy factor. T is equal to 1 for controlled areas, and may be less than 1 for environs (see Table 8-43).^c Approximately.^d Pounds per square foot computed from millimeters of lead or inches of concrete and a concrete density of 2.35 g/cm³ (147 lb/ft³).

Table 8-50. Shielding Requirements for Busy 150-kv Therapeutic Installations ^a

Type of barrier	UT ^b	Distance from tube to occupied area											
		5 ft (1.52 m)				7 ft (2.13 m)				10 ft (3.05 m)			
		Lead		Concrete		Lead		Concrete		Lead		Concrete	
		Mm	In. ^c	Pst ^d	In.	Mm	In. ^c	Pst ^d	In.	Mm	In. ^c	Pst ^d	In.
For controlled areas:	Primary	1	3.1	3/4	7.2	10.4	127	2.9	3/4	6.8	10.0	123	2.5
			2.5	3/32	5.8	8.8	108	2.3	3/32	5.4	8.2	100	2.0
			2.0	3/32	4.7	7.2	88	1.8	1/16	4.2	6.5	90	1.5
Secondary	1	1.7	1/16	4.0	6.1	75	1.5	1/16	3.5	5.4	66	1.2	3/16
For environs:	Primary	1	4.1	3/32	9.6	13.3	163	3.9	3/32	9.1	12.8	157	3.5
			3.5	3/32	8.2	11.6	142	3.3	1/4	7.7	11.1	136	3.0
			3.0	1/4	7.0	10.2	125	2.8	1/4	6.5	9.7	119	2.4
			2.4	3/32	5.6	8.5	104	2.2	3/32	5.1	7.8	96	1.8
Secondary	1	2.6	3/32	6.1	9.1	111	2.4	3/32	5.6	8.5	104	2.0	3/32
			2.0	3/32	4.7	7.2	88	1.8	1/16	4.2	6.5	90	1.5
			1.4	1/16	3.3	5.1	62	1.2	1/16	2.8	4.3	53	0.9

^a For design levels of 100 mr/week for controlled areas and 10 mr/week for environs, and for a weekly work load of 4,000 ma-min at 150 kvp.^b U = use factor; T = occupancy factor. T is equal to 1 for controlled areas, and may be less than 1 for environs (see Table 8-43).^c Approximately.^d Pounds per square foot computed from millimeters of lead or inches of concrete and a concrete density of 2.35 g/cm³ (147 lb./ft³).

RADIATION ATTENUATION DATA

Table 8-51. Shielding Requirements for Busy 200-kv Therapeutic Installations ^a

Type of barrier	U ^{Tb}	Distance from tube to occupied area																								
		5 ft (1.52 m)				7 ft (2.13 m)				10 ft (3.05 m)				14 ft (4.26 m)				20 ft (6.10 m)								
		Lead		Concrete		Lead		Concrete		Lead		Concrete		Lead		Concrete		Lead		Concrete						
		Mm	In. ^c	Psf ^d	In.	Mm	In. ^c	Psf ^d	In.	Mm	In. ^c	Psf ^d	In.	Mm	In. ^c	Psf ^d	In.	Mm	In. ^c	Psf ^d	In.					
For controlled areas:	1	6.4	3/4	14.9	16.7	205	5.9	3/4	13.7	15.8	194	5.4	7/32	12.6	14.8	181	4.9	3/16	11.4	13.7	168	4.4	3/16	10.3	12.7	156
	3/4	5.4	7/32	12.6	14.8	181	4.9	3/16	11.4	13.7	168	4.4	3/16	10.3	12.7	156	3.9	3/32	9.1	11.6	142	3.4	3/4	7.9	10.5	129
	3/16	4.4	3/16	10.3	12.7	156	3.9	3/32	9.1	11.6	142	3.4	3/4	7.9	10.5	129	3.0	3/4	7.0	9.6	118	2.6	3/32	6.1	8.5	104
Secondary	1	4.1	5/32	9.6	12.1	148	3.6	5/32	8.4	11.0	135	3.1	3/4	7.2	9.8	20	2.6	3/32	6.1	8.5	104	2.2	3/32	5.1	7.4	91
	For environs:																									
	1	8.1	1 1/32	18.9	19.9	244	7.6	5/16	17.7	19.0	233	7.1	5/32	16.5	18.0	221	6.6	5/32	15.4	17.1	209	6.1	3/4	14.2	16.2	198
Primary	3/4	7.1	9/32	16.5	18.0	221	6.6	5/32	15.4	17.1	209	6.1	3/4	14.2	16.2	198	5.6	7/32	13.0	15.2	186	5.1	7/32	11.9	14.2	174
	3/16	6.1	3/4	14.2	16.2	198	5.6	7/32	13.0	15.2	186	5.1	7/32	11.9	14.2	174	4.6	3/16	10.7	13.1	160	4.1	5/32	9.6	12.1	148
	3/64	5.1	7/32	11.9	14.2	174	4.6	3/16	10.7	13.1	160	4.1	5/32	9.6	12.1	148	3.6	3/32	8.4	11.0	135	3.2	3/4	7.5	10.0	123
Secondary	3/32	4.1	5/32	9.6	12.1	148	3.6	3/32	8.4	11.0	135	3.2	3/4	7.5	10.0	123	2.7	3/4	6.3	8.8	108	2.3	3/32	5.4	7.7	94
	1	5.6	7/32	13.0	15.2	186	5.1	7/32	11.9	14.2	174	4.6	3/16	10.7	13.1	160	4.1	5/32	9.6	12.1	148	3.6	5/32	8.4	11.0	135
	3/4	4.6	3/16	10.7	13.1	160	4.1	5/32	9.6	12.1	148	3.6	3/32	8.4	11.0	135	3.2	3/4	7.5	10.0	123	2.7	3/4	6.3	8.8	108
	3/16	3.6	5/32	8.4	11.0	135	3.2	3/4	7.5	10.0	123	2.7	3/4	6.3	8.8	108	2.3	3/32	5.4	7.7	94	1.7	3/16	4.0	6.0	74

^a For design levels of 100 mr/week for controlled areas and 10 mr/week for environs, and for a weekly work load of 40,000 ma-min at 200 kvp. Density of concrete 147 lb./ft.³.^b U = use factor; T = occupancy factor. T is equal to 1 for controlled areas, and may be less than 1 for environs (see Table 8-43).^c Approximately.^d Pounds per square foot computed from millimeters of lead or inches of concrete and a concrete density of 2.35 g/cm³ (147 lb./ft.³).

Table 8-52. Shielding Requirements for Busy 250-kv Therapeutic Installations ^a

Type of barrier	U ^b	Distance from tube to occupied area											
		5 ft (1.52 m)				7 ft (2.13 m)				10 ft (3.05 m)			
		Lead		Concrete		Lead		Concrete		Lead		Concrete	
		Mm	In. ^c	Psg ^d	In.	Mm	In. ^c	Psg ^d	In.	Mm	In. ^c	Psg ^d	In.
For controlled areas:	Primary.....	11.3	15/32	26.4	19.3	236	10.4	7/16	24.3	18.0	221	22.2	16.8
	1/4	9.5	3/8	22.2	16.8	206	8.6	15/32	20.1	15.6	191	17.7	14.4
	1/16	7.7	5/16	18.0	14.4	176	6.8	9/32	15.9	13.2	162	6.0	12.0
Secondary.....	1	6.4	1/4	14.9	12.6	154	5.7	7/32	13.3	11.5	141	5.0	7/32
	1/4	12.1	1/2	28.2	20.4	250	11.2	15/32	26.1	19.2	235	10.3	13/32
	1/16	10.3	15/32	24.0	18.0	221	9.4	3/8	21.9	16.7	205	8.5	15/32
For environs:	Primary.....	13.9	9/16	32.4	22.9	281	13.0	17/32	30.3	21.6	265	12.1	15/32
	1/4	12.1	1/2	28.2	20.4	250	11.2	15/32	26.1	19.2	235	10.3	13/32
	1/16	10.3	15/32	24.0	18.0	221	9.4	3/8	21.9	16.7	205	8.5	15/32
Secondary.....	1	8.8	15/32	20.5	16.0	196	8.0	5/16	18.7	14.8	181	7.3	9/32
	1/4	7.3	9/32	17.0	13.8	169	6.5	3/4	15.2	12.7	156	5.8	3/4
	1/16	5.8	1/4	13.5	11.7	143	5.0	7/32	11.7	10.5	129	4.4	3/8

^a For design levels of 100 mr/week for controlled areas and 10 mr/week for environs, and for a weekly work load of 40,000 ma-min at 250 kvp. Density of concrete 147 lb/ft³.^b U = use factor; T = occupancy factor. T is equal to 1 for controlled areas, and may be less than 1 for environs (see Table 8-43).^c Approximately.^d Pounds per square foot computed from millimeters of lead or inches of concrete and a concrete density of 2.35 g/cm³ (147 lb/ft³).

Table 8-53. Industrial X-ray Shielding Requirements for Controlled Areas ^a

WUT ^b			Distance from tube to occupied area, ft												
Poten- tial, kv	Approximate ϵ HVL, mm	Approximate ϵ TVL, mm	Type of barrier	5	7	10	14	20	28	40					
100	0.2	0.7	Prim.	3.2	2.9	2.7	2.4	2.2	1.9	1.7	1.4	1.2	1.0	0.8	0.8
150	0.3	1.0	Sec.	2.1	1.8	1.6	1.4	1.2	1.0	0.8	0.6	0.4	0.2	0.1	0.1
200	0.5	1.7	Prim.	4.1	3.8	3.5	3.3	3.0	2.7	2.4	2.1	1.8	1.6	1.3	1.3
250	0.9	3.0	Sec.	2.6	2.3	2.0	1.8	1.5	1.2	0.9	0.7	0.5	0.3	0.1	0.1
300	1.7	5.6	Prim.	6.4	5.9	5.4	4.9	4.4	3.9	3.4	3.0	2.6	2.3	2.0	2.0
			Sec.	4.1	3.7	3.1	2.6	2.2	1.8	1.4	1.1	0.8	0.5	0.2	0.2
			Prim.	11.3	10.4	9.5	8.6	7.7	6.8	6.0	5.2	4.4	3.6	2.8	2.8
			Sec.	6.4	5.7	5.0	4.2	3.5	2.8	2.2	1.6	1.2	0.8	0.3	0.3
			Prim.	21.6	19.8	18.0	16.3	14.6	13.0	11.4	9.8	8.3	6.8	5.5	5.5
			Sec.	12.0	10.4	8.9	7.5	6.3	5.2	4.2	3.2	2.3	1.5	0.6	0.6

Table 8-54. Industrial X-ray Shielding Requirements for Controlled Areas ^a

WUT ^b				Distance from tube to occupied area, ft										
Poten- tial, kv	Approximate ^c HVL, in.	Approximate ^c TVL, in.	Type of barrier	Thickness of concrete (density 147 lb/ft ³) ^d , in.										
100	0.6	2.0	Prim. Sec.	9.2 6.6 13.3	8.6 5.8 12.4	8.0 5.2 11.5	7.3 4.6 10.6	6.6 4.0 9.7	6.0 3.4 8.9	5.5 2.8 8.1	4.6 2.3 7.2	4.0 1.8 6.5	3.3 1.3 5.8	2.8 0.8 5.1
150	0.9	3.0	Prim. Sec.	9.1 16.9	8.2 15.8	7.3 14.8	6.4 13.7	5.5 12.7	4.6 11.6	3.7 10.6	2.9 9.6	2.2 8.5	1.7 7.5	1.2 6.5
200	1.1	3.6	Prim. Sec.	12.1 19.3	10.9 18.0	9.8 16.8	8.7 15.6	7.6 14.4	6.6 13.2	5.6 12.0	4.6 10.8	3.6 9.6	2.6 8.4	1.6 7.2
250	1.1+	3.8	Prim. Sec.	12.6 21.7	11.4 20.5	10.3 19.3	9.2 18.1	8.1 16.9	7.0 15.7	6.0 14.5	5.0 13.3	4.0 12.1	3.0 10.9	2.0 9.7
300	1.2	4.0	Prim. Sec.	13.4 12.2	12.2 11.0	11.0 9.8	9.8 8.8	8.8 7.8	7.8 6.8	6.8 5.8	5.8 4.8	4.8 3.8	3.8 2.8	2.8 0.8

^a For a design level of 100 mr/week. Add one tenth-value layer (TVL) for environs to reduce radiation to 10 mr/week.^b W = work load in ma-min/week; U = use factor; T = occupancy factor. T is equal to 1 for controlled areas and may be less than 1 for environs (see Table 8-43).^c These values are obtained at high filtration.^d For other densities ρ , the tabular values should be multiplied by $147/\rho$.

Table 8-55. Average Shielding Requirements for 1-million-volt X-ray Installations ^a

Type of barrier	UT ^b	Distance from tube to occupied area																								
		7 ft (2.14 m)				10 ft (3.05 m)				14 ft (4.28 m)				20 ft (6.10 m)				30 ft (9.15 m)								
		Lead		Concrete		Lead		Concrete		Lead		Concrete		Lead		Concrete		Lead		Concrete						
		Mm	In. ϵ	Psf ^d	In.	Psf ^d	Mm	In. ϵ	Psf ^d	In.	Psf ^d	Mm	In. ϵ	Psf ^d	In.	Psf ^d	Mm	In. ϵ	Psf ^d	In.	Psf ^d					
For controlled areas: Primary.....	1	130	5.1	300	32	390	120	4.7	290	30	370	115	4.5	270	28½	350	105	4.1	240	26½	330	100	3.9	230	25	310
	½	115	4.5	270	28½	350	105	4.1	240	28½	330	100	3.9	230	25	310	90	3.5	200	23	280	85	3.3	200	21½	260
	⅓	100	3.9	230	25	310	90	3.5	210	23	290	85	3.3	200	21½	260	75	3.0	180	19½	240	70	2.8	170	18	220
For envious: Primary.....	1	160	6.3	370	38	470	145	5.7	330	36	440	140	5.5	320	34½	420	130	5.1	300	32½	400	125	4.9	290	31	380
	½	140	5.5	320	34½	430	130	5.1	300	32½	400	125	4.9	290	31	380	115	4.5	270	29	360	110	4.3	250	27½	340
	⅓	125	4.9	290	31	380	115	4.5	270	29	360	110	4.3	250	27½	340	100	3.9	230	25½	310	90	3.5	210	24	290
	⅓	110	4.3	250	27½	340	100	3.9	230	25½	310	90	3.5	210	24	290	80	3.2	190	22	270	75	3.0	170	20½	250
	⅓	90	3.5	210	24	290	80	3.2	190	22	270	75	3.0	170	20½	250	65	2.6	150	18½	230	60	2.4	140	17	210

^a For design levels of 100 mr/week for controlled areas and 10 mr/week for environs, and for a weekly work load of 4,000 ma-min.

^b U = use factor; T = occupancy factor. T is equal to 1 for controlled areas, and may be less than 1 for environs (see Table 8-43).

^c Approximately.

^d Pounds per square foot computed from millimeters of lead or inches of concrete and a concrete density of 2.35 g/cm³ (147 lb./ft.³)

Table 8-56. Average Shielding Requirements for 2-million-volt X-ray Installations ^a

Type of barrier	U/T ^b	Distance from tube to occupied area																								
		7 ft (2.14 m)			10 ft (3.05 m)			14 ft (4.28 m)			20 ft (6.10 m)			30 ft (9.15 m)												
		Lead		Concrete	Lead		Concrete	Lead		Concrete	Lead		Concrete	Lead		Concrete										
		Mm	In. ^c	Psf ^d	In.	Psf ^d	Mm	In. ^c	Psf ^d	In.	Psf ^d	Mm	In. ^c	Psf ^d	In.	Psf ^d	Mm	In. ^c	Psf ^d	In.	Psf ^d					
For controlled areas:	1	235	9.3	550	50	610	225	8.9	520	47	580	210	8.3	490	45	550	200	7.9	470	42	520	185	7.3	430	40	490
	1/4	210	8.3	490	45	550	200	7.9	470	42	520	185	7.3	430	40	490	175	6.9	410	37	460	160	6.3	370	35	430
	1/16	185	7.3	430	40	490	175	6.9	410	37	460	160	6.3	370	35	430	150	5.9	350	32	390	135	5.3	320	30	370
For environs:	1	280	11.1	650	58	710	265	10.5	620	55	670	255	10.1	590	53	650	240	9.5	560	50	610	230	9.1	540	48	590
	1/4	255	10.1	590	53	650	240	9.5	560	50	610	235	9.1	540	48	590	215	8.5	500	45	550	205	8.1	480	43	530
	1/16	230	9.1	540	48	590	215	8.5	500	45	550	205	8.1	480	43	530	190	7.5	440	40	490	180	7.1	420	38	470
	1/64	205	8.1	480	43	530	190	7.5	440	40	490	180	7.1	420	38	470	165	6.5	380	35	430	155	6.1	360	33	400
	1/256	180	7.1	420	38	470	165	6.5	390	35	430	155	6.1	360	33	400	140	5.5	330	30	370	130	5.1	300	28	340

^a For design levels of 100 mr/week for controlled areas and 10 mr/week for environs, and for a weekly work load of 2,000 ma-min.^b U = use factor; T = occupancy factor. T is equal to 1 for controlled areas, and may be less than 1 for environs (see Table 8-43).^c Approximately.^d Pounds per square foot computed from millimeters of lead or inches of concrete and a concrete density of 2.35 g/cm³ (147 lb/ft³).

Table 8-57. Primary Barrier Thicknesses and Weight of Concrete for Multimegavolt Installations

	Tube potential	WUT	Concrete thickness required for WUT^a and distances from target to occupied area of											
			Ft		M		Ft		M		Ft		M	
			Ft	M	Ft	M	Ft	M	Ft	M	Ft	M	Ft	M
Controlled area	6 10 20 40-180	1,600	10	3.05	20	6.10	40	12.2	80	24.4	80			
		400	5	1.52	10		20		40		40			
		100			5		10		20		20		80	
		25					5		10		20		40	
		6							5		10		40	
Environs	6 10 20 40-180	1½							5		5		20	
			In.	Psf ^b	In.	Psf ^b	In.	Psf ^b	In.	Psf ^b	In.	Psf ^b	In.	Psf ^b
			42	510	34	420	26	320	18	220	11	135	2	25
			49	600	40	490	31	380	21	260	12	145	3	40
			59	720	48	590	37	450	26	320	15	185	4	50
Environs	6 10 20 40-180		61	750	50	610	40	490	29	360	17	210	4	50
			55	670	48	590	40	490	32	390	23	280	15	180
			64	780	55	670	46	560	37	450	28	340	18	220
			77	940	66	810	56	690	45	550	34	420	23	280
			79	970	68	830	58	710	47	580	36	440	25	310

^a W = weekly work load in esu/cm² (lucite) at 1 m; U = use factor; T = occupancy factor. T is equal to 1 for controlled areas, and may be less than 1 for environs (see Table 8-43).

^b Assuming a density of 147 lb/ft³. Psf = pounds per square foot of wall area.

Table 8-58. Primary Barrier Thicknesses and Weight of Lead for Multimegavolt Installations

	WUT	Lead thickness required for WUT^a and distances from target to occupied area of											
		Ft			M			Ft			M		
		Ft	M	Ft	M	Ft	M	Ft	M	Ft	M	Ft	M
Tube potential	1,600	10	3.05	20	6.10	40	12.2	80	24.4				
	400	5	1.52	10		20		40		80			
	100			5		10		20		40		80	
	25					5		10		20		40	
	6							5		10		20	
	$1\frac{1}{2}$									5		10	
Mev		In.	Psf ^b	In.	Psf ^b	In.	Psf ^b	In.	Psf ^b	In.	Psf ^b	In.	Psf ^b
Controlled area	10	6.7	400	5.5	330	4.2	250	2.9	170	1.7	100	0.4	24
	86-176	5.7	340	4.6	270	3.6	210	2.5	150	1.4	80	0.4	24
Environments	10	8.9	530	7.6	450	6.3	370	5.1	300	3.8	230	2.5	150
	86-176	7.5	440	6.4	380	5.3	310	4.3	250	3.2	190	2.1	120
												1.3	80
												1.1	70

^a W = weekly work load in esu/cm³ (lucite) at 1 m; U = use factor; T = occupancy factor. T is equal to 1 for controlled areas, and may be less than 1 for environs (see Table 8-43).

^b psf = pounds per square foot of wall area.

Table 8-59. Cobalt-60 Shielding Requirements for Controlled Areas ^a

WUT ^b	Curies ^c (approx.)	Distance from source to occupied areas, ft									
		5	7	10	14	20	28	40	40	40	40
80,000	2,000										
40,000	1,000		5	7	10	14	20	28	40	40	40
20,000	500			5	7	10	14	20	28	28	28
10,000	250				5	7	10	14	20	20	20
5,000						5	7	10	14	14	14
2,500							5	7	10	10	10
1,250								5	7	7	7
625									5	5	5
310										5	7

Type of barrier	HVL, cm (approx.)	TVL, cm (approx.)	Thickness of lead, cm									
			22.5	21.3	20.1	18.9	17.7	16.5	15.3	14.1	12.9	11.7
Primary.....	1.20	4.0										
Secondary:												
Leakage ^d 0.1 %	1.20	4.0	10.5	9.3	8.2	7.1	6.0	4.7	3.4	2.0	0.6	0
0.05 %	1.20	4.0	9.3	8.2	7.1	6.0	4.7	3.4	2.0	0.6	0	0
Scatter ^e												
30°	1.02	3.40	12.15	11.10	10.10	9.05	8.00	7.00	6.00	4.95	3.95	2.95
45°	0.87	2.90	9.65	8.75	7.90	7.00	6.15	5.25	4.35	3.50	2.65	1.75
60°	0.75	2.50	7.65	6.90	6.15	5.40	4.65	3.90	3.15	2.4	1.65	0.9
90°	0.43	1.45	3.60	3.15	2.70	2.30	1.85	1.40	1.00	0.60	0.25	0.03
120°	0.20	0.65	1.45	1.25	1.05	0.85	0.65	0.45	0.30	0.15	0.05	0
150°	0.14	0.45	0.90	0.75	0.65	0.50	0.40	0.25	0.15	0.08	0.02	0

^a For a weekly design level of 100 mr; add one tenth-value layer (TVL) for regions in the environs to reduce radiation to 10 mr/week.^b W = work load in r/week at 1 m; U = use factor; T = occupancy factor.^c Assumes use factor U and occupancy factor T are equal to one.^d Refers to leakage radiation of source housing; may be ignored if less than 2.5 mr/hr at 1 m in "on" position.^e For large field (20 cm diam) and a source-skin distance of 40 to 60 cm.

Table 8-60. Cobalt-60 Shielding Requirements for Controlled Areas ^a

WUT ^b	Curies ^c (approx.)	Distance from source to occupied areas, ft									
		5	7	10	14	20	28	40	40	40	40
80,000	2,000	5	7	10	14	20	28	40	40	40	40
40,000	1,000		5	7	10	14	20	28	28	28	28
20,000	500			5	7	10	14	20	20	20	20
10,000	250				5	7	10	14	14	14	14
5,000						5	7	10	10	10	10
2,500							5	7	7	7	7
1,250								5	5	5	5
625											
310											

Type of barrier	HVL, in. (approx.)	TVL, in. (approx.)	Thickness of concrete (density 147 lb./ft. ³), in.									
			47.5	45.1	42.7	40.3	37.8	35.4	32.9	30.5	28.0	25.6
Primary.....	2.6	8.6	23.1	20.7	18.3	15.6	12.9	10.2	7.3	4.4	1.4	0
Secondary:	2.6	8.6	20.7	18.3	15.6	12.9	10.2	7.3	4.4	1.4	0	0
Leakage ^d	0.1 %											
	0.05 %											
Scatter ^e	30°	8.2	30.6	28.1	25.7	23.2	20.6	18.2	15.8	13.3	10.9	8.4
	45°	8.0	27.0	24.6	22.2	19.8	17.4	15.0	12.6	10.2	7.8	5.4
	60°	7.6	24.0	21.7	19.4	17.1	14.8	12.3	10.0	7.7	5.4	3.0
	90°	6.2	16.9	15.0	13.2	11.4	9.6	7.8	6.0	4.1	2.1	0.1
	120°	5.8	15.0	13.3	11.6	9.9	8.2	6.5	4.7	3.0	1.2	0
	150°	5.0	9.4	7.9	6.4	4.9	3.4	1.8	0.3	0	0	0

^a For a weekly design level of 100 mr; add one tenth-value layer (TVL) for regions in the environs to reduce radiation to 10 mr/week.^b W = work load in r/week at 1 m; U = use factor; T = occupancy factor.^c Assumes use factor U and occupancy factor T are equal to one.^d Refers to leakage radiation of source housing; may be ignored if less than 2.5 mr/hr at 1 m in "on" position.^e For large field (20 cm diam) and a source-skin distance of 40 to 60 cm.

Table 8-61. Cesium-137 Shielding Requirements for Controlled Areas ^a

WUT ^b	Curies ^c (approx.)	Distance from source to occupied areas, ft												
		5	7	10	14	20	28	40	40	40	28	20	14	
24,000	2,000													
12,000	1,000													
6,000	500													
3,000	250													
1,500	125													
750														
375														
Type of barrier	HVL, cm (approx.)	TVL, cm (approx.)	Thickness of lead, cm											
	0.65	2.1	10.5	9.9	9.3	8.6	8.0	7.4	6.7	6.1	5.5	4.8	4.1	
Primary.....														
Secondary:														
Leakage ^d	0.1 %	2.1	4.2	3.5	2.9	2.3	1.6	1.0	0.4	0	0	0	0	
	0.05%	2.1	3.5	2.9	2.3	1.6	1.0	0.4	0	0	0	0	0	
Scatter ^e	35°	1.5	5.3	4.9	4.4	3.9	3.5	3.0	2.6	2.2	1.7	1.3	0.8	
	56°	1.3	4.1	3.7	3.3	2.9	2.5	2.1	1.7	1.4	1.0	0.7	0.4	
	90°	0.7	2.0	1.8	1.6	1.3	1.1	0.9	0.7	0.5	0.4	0.2	0.1	
	119°	0.4	1.0	0.9	0.8	0.7	0.6	0.5	0.3	0.2	0.2	0.1	0.03	

^a For a weekly design level of 100 mr; add one tenth-value layer (TVL) for regions in the environs to reduce to 10 mr/week.^b W = work load in r/week at 1 m; U = use factor; T = occupancy factor.^c Assumes use factor U and occupancy factor T are equal to one.^d Refers to leakage radiation of source housing; may be ignored if less than 2.5 mr/hr at 1 m in "on" position.^e For large field (20 cm diam) and a source-scatterer distance of 50 cm.

Table 8-62. Cesium-137 Shielding Requirements for Controlled Areas ^a

WUT ^b	Curies ^c (approx.)	Distance from source to occupied areas, ft											
		5	7	10	14	20	28	40	40	40	40	40	40
24,000	2,000	5	7	10	14	20	28	40	40	40	40	40	40
12,000	1,000		5	7	10	14	20	28	28	28	28	28	28
6,000	500			5	7	10	14	20	20	20	20	20	20
3,000	250				5	7	10	14	14	14	14	14	14
1,500	125					5	7	10	10	10	10	10	10
750							5	7	7	7	7	7	7
375								5	5	5	5	5	5

Type of barrier	HVL, cm (approx.)	TVL, cm (approx.)	Thickness of concrete, in.											
			34.1	32.2	30.2	28.3	26.3	24.4	22.4	20.6	18.7	16.8	14.8	
Primary.....	1.9	6.2	14.8	12.9	11.0	9.1	7.0	4.9	2.0	0	0	0	0	
Secondary:			12.9	11.0	9.1	7.0	4.9	2.8	0	0	0	0	0	
Leakage ^d	0.1 %	6.2	23.4	21.6	19.8	18.0	16.2	14.4	12.5	10.7	8.8	7.0	5.1	
Scatter ^e	35°	6.1	18.8	17.3	15.8	14.3	12.8	11.3	9.8	8.4	6.9	5.5	4.0	
	56°	4.9	16.3	14.9	13.5	12.0	10.6	9.2	7.8	6.4	5.0	3.7	1.9	
	90°	4.7	14.9	13.6	12.3	11.0	9.6	8.3	7.0	5.7	4.3	3.0	1.3	
	119°	4.4												

^a For a weekly design level of 100 mr; add one tenth-value layer (TVL) for regions in the environs to reduce to 10 mr/week.^b W = work load in r/week at 1 m; U = use factor; T = occupancy factor.^c Assumes use factor U and occupancy factor T are equal to one.^d Refers to leakage radiation of source housing; may be ignored if less than 2.5 mr/hr at 1 m in "on" position.^e For large field (20 cm diam) and a source-scatterer distance of 50 cm.

Section 9

LABORATORY DESIGN

By

G. W. MORGAN, *Chief, Isotopes Extension, Division of Civilian Application,
U.S. Atomic Energy Commission, Washington, D.C.*

Location.....	9-2
General Features.....	9-3
Furniture.....	9-4
Ventilation.....	9-7
Plumbing.....	9-7
Laboratory Layout.....	9-8

LABORATORY DESIGN

G. W. Morgan

See page 9-11 for General References.

Laboratories designed for radioisotope work require **specialized features** to (1) protect personnel from exposure to radiation and from bodily intake of the radio-materials and (2) prevent invalidation of radioassays and radiation measurements. The required degree of modification from ordinary laboratories depends upon the type and magnitude of the radioisotope program, i.e., things such as (1) type and level of activity, (2) chemical and physical form of the radiomaterials, (3) type of manipulation, (4) number of people participating in the program, and (5) volume of work.

Added **space** is necessary so that procedures can be segregated and that specialized equipment can be used. Space requirements will vary from a small room or even a portion of a room to whole buildings. For the typical research or industrial program, however, a two- or three-room laboratory is usually adequate. Changes in type and magnitude of programs should be anticipated and provisions made for modifications to meet future needs. This section is limited to the basic principles of design and illustrations of typical research, industrial, and medical programs.

LOCATION

The importance of **location** depends upon the type and magnitude of the program. For the typical program, isolated locations are not essential, but there are **special considerations** which are necessary. Considerations given to determining the location should include (1) shielding requirements, (2) problems of transporting shielded containers, (3) requirements for control of effluents, (4) proximity to supporting facilities, (5) technical requirements to assure valid measurements, and (6) storage requirements.

Location on the ground floor or in the basement is advantageous from the standpoint of **shielding** in that the earth can often be utilized for this purpose and that floors can easily be reinforced to support any added weight. Likewise, this location will simplify transfer problems when heavy shielding is used. Also, this location permits direct drainage of **liquid effluents** into the main sewer lines or hold-up tanks. Conversely, location on the top floor simplifies problems associated with **gaseous or air-borne exhaust** from fume hoods. This may be an important item when the laboratory is installed in an existing building where ductwork may have to be extended through floors above the laboratory. However, other reasons, such

as economy or necessity of locating near already established facilities, may outweigh both these considerations, particularly when an existing laboratory is to be modified for radioactive work.

From the technical standpoint, counting equipment must be protected from stray radiation such as that arising from the storage or use of radioisotopes or from the operation of X-ray machines and accelerators. Therefore, it is undesirable to locate a radioisotope laboratory near X-ray machines, gamma irradiation facilities, or accelerators.

The radioisotope laboratory should be convenient to supporting laboratories and facilities, and it should be in an area where the traffic and access of other personnel can be controlled to a degree compatible with the potential hazards. Usually it is preferable to group the rooms in the radioisotope laboratory together to permit more efficient use of facilities and personnel and to simplify radiation-hazard-control procedures. But, in some instances, it may be desirable to set up a centralized "hot" or "semihot" laboratory for receiving and dispensing radioisotopes and for performing operations involving potentially significant hazards. With this arrangement, other laboratories in the institution can be restricted to handling low-level activity and thus will require only minor modifications.

GENERAL FEATURES

Facilities should be designed to provide maximum **control of radioactive materials**. This is accomplished by special design features which minimize the spread of contamination and simplify decontamination procedures. Avoid dirt catchers. Sharp corners, cracks, ledges, exposed pipes, and loose wiring should be eliminated where possible. Air contaminants must either be confined in a closed system or be controlled by adequate ventilating systems such as hoods. External exposure should be minimized by proper use of shielding and remote-handling equipment.

Laboratory surfaces should be easily decontaminated. The surface of **floors, walls, and ceilings** should be nonporous and washable. Those constructed of wood, concrete, or cinder block should be protected by **paints** from which the materials can be removed by simple wash-and-scrub procedures. Though almost any good paint will provide some protection for surfaces, it is desirable to use paints of proved value to cover surfaces such as bench tops or the interior of hoods. Usually several coats should be applied, and the base coat should be capable of penetrating and sealing the pores of the surface. For extensive radioisotope programs, walls behind work benches, bench tops, and hood interiors should be painted at critical points with special strippable coatings. Concrete and wood floors should be covered with linoleum or one of the tiles such as vinyl, asphalt, or rubber. For a simple radioisotope program, linoleum laid in the conventional manner should be adequate. But where there is a probability of significant spillage, block tiles are desirable because of the ease of removal and repair of small areas otherwise difficult to decontaminate. **Wax** coatings protect the floor and working surfaces, fill the cracks, and simplify decontaminating procedures.

When gamma-ray emitters are used, floors, hoods, and bench structure must be sturdy enough to sustain the required weight of shielding. Barricades used in

hoods may weigh 1,000 lb or more; therefore, hoods, benches, and floors may require **special structural support**.

FURNITURE

Laboratory benches should provide the usual service outlets. Vacuum lines should have traps to catch radioactive liquids that might be accidentally drawn into them. Cup sinks are desirable to minimize accumulation of extraneous materials and simplify housekeeping. Stainless-steel tops are preferable, but for simple programs involving small contamination potentials, ordinary laboratory bench tops may be adequate if protected by appropriate paints. For additional control of spills, bench surfaces should be further protected by absorbent paper, and trays should be utilized to the maximum. Slightly wider benches may be desired where operation requires considerable remote-control equipment or shielding.

Stainless-steel **laboratory sinks** are desirable in most cases. Enamel sinks are adequate except where operations are likely to produce flakage or pitting. For programs requiring considerable use of sinks, double drain boards and drain racks are required to segregate and dry glassware. Corners should be rounded for easy cleaning. In programs where contamination could be significant the faucet should be foot- or knee-operated.

Laboratory shielding requirements vary with the type of radiation emitted by the radioisotope, the level of activity, and the type of program. Shielding is no problem for **alpha** emitters since they are incapable of penetrating the human skin. Most **beta** particles are completely stopped by $\frac{1}{4}$ in. of plastic or wood but have a range in air up to several meters. On the other hand, very thick shields of concrete, iron, or lead may be required for protection from **gamma** emitters, the thickness depending upon the energy of the radiation and the level of activity. Shields are required for work with most radioactive materials in greater than tracer quantities.

Shielding may be either portable or permanent and may take the form of a "close shield" or of a "barrier shield." The **close shield** is essentially a container of sufficient thickness to reduce the radiation to the desired level. This type of shielding offers the most efficient use of shield material as it provides maximum thickness with minimum mass. The **barrier shield** is essentially a wall or enclosure for shielding an area in which manipulations can be performed. It permits a greater degree of flexibility in setting up equipment but generally entails greater expense as it requires more shielding material in order to attain the same thickness as the close shield.

The typical radioisotope laboratory usually relies heavily on **portable shielding**. Portable close shielding is most applicable to the less penetrating radiation and to portable storage containers. For example, a lead brick with holes bored part way through it makes a good test-tube rack for some applications involving low and intermediate levels. Lead or iron bricks offer flexibility and mobility in constructing various sizes of barricade shields. For the typical program, permanent shielding is usually restricted to storage areas and facilities for routine types of radioisotope work, the former usually utilizing the close-shield principles and the latter frequently the barricade shield.

Hoods are extensively used in radiochemical laboratories to perform a dual function, removing contaminated air and providing a confined working space for controlling surface contamination. Hoods should be designed to maintain an adequate and uniform rate of air flow. Especially designed hoods such as that shown in Fig. 9-1 are preferable when significant contamination is probable.

Selecting a hood location in the room is influenced by feasible duct location, by architectural support available to support shielding which may be required in the hood, by location of heating and ventilation systems, and by the relative location of other radiochemical operations. Excessive lengths and bends in the ductwork increase resistance in the system, thereby requiring increased power of the blower motor. The exhaust blower should be located near the roof to maintain a negative air pressure in as much as possible of the exhaust duct. The blower motor should be mounted outside the air duct to prevent it from becoming contaminated. The optimum height of an exhaust duct above the roof is dependent upon a number of factors which require an evaluation of each individual situation.

If the type of program requires filtration of the hood effluent, simple mechanical **filters** are most easily utilized. Glass wool may be used for filtering out coarse particulates while special filters are required for removal of smaller particles. Filter boxes should be designed to permit accessibility for the purpose of changing filters without undue contamination of the area. A permanent manometer installed across the filter is helpful in indicating the need for filter change. Filter resistance must be considered in hood-system design, especially in selecting a blower motor with adequate power. Prefiltering of room air supply is sometimes desirable to decrease the particulate loading of the more expensive hood filters (see also Sec. 22).

The air-flow characteristics of hoods are affected by air drafts in the vicinity of the hood face. For this reason hoods should not be placed near windows, ventilation systems, heaters, doors, or other sources of strong air currents. The air withdrawn from the room by the hood will affect the general air circulation of the laboratory and must be considered in establishing a desirable pattern of air movement.

Operation of two or more hoods with one blower will require attention to the necessity of maintaining a balanced system, with each hood having an adequate air flow. This is especially true with high-resistance systems utilizing filters. When more than one hood with independent blowers is located in the same laboratory room, it is desirable to control the blowers by a single electrical switch in order to avoid a reverse flow of air through a nonoperating hood which would contaminate the area.

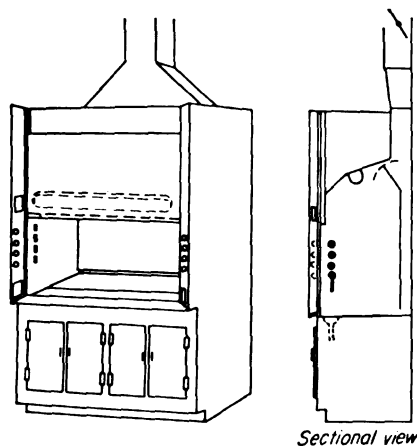


FIG. 9-1. Radioisotope hood.

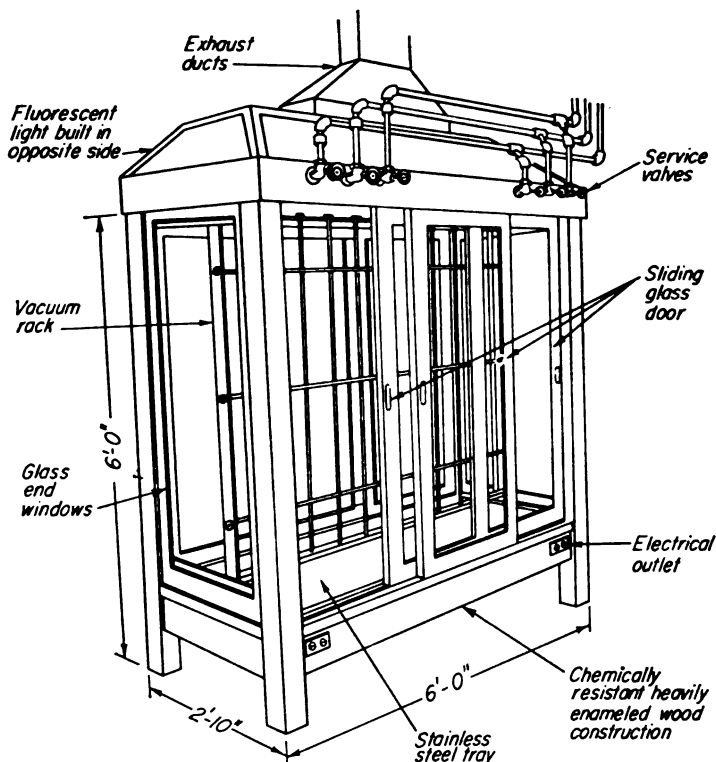


FIG. 9-2. California hood.

The **California hood** is often used in organic synthesis and other procedures involving vacuum-line equipment. It is particularly adaptable for operations involving weak beta emitters, such as tritium, Carbon-14, or Sulfur-35. The California hood is a cabinet-type structure especially designed to house this type of equipment (see Fig. 9-2).

Glove boxes (or **dry boxes**) are often utilized when radioactive materials are used in open vessels, especially where materials emit alpha or low-energy beta particles. They are essentially small enclosures for localizing contamination. Operations are performed by use of rubber gloves or mechanical manipulators mounted within the structure. A small blower operated by a fractional-horsepower motor is used to maintain a slightly negative pressure within the box. The exhausted air may be filtered or trapped before being discharged into a manifold or standard hood. Glove boxes are particularly adaptable for use with alpha emitters since operations are not hampered by shielding. When beta emitters are used except those with very low energy, shielding is required. For the typical institution and industrial-research program, glove boxes as illustrated in Fig. 9-3 are adequate. Elaborate glove boxes have been developed for use with very hazardous materials and for laboratories developed for maximum use of these facilities in involved programs.

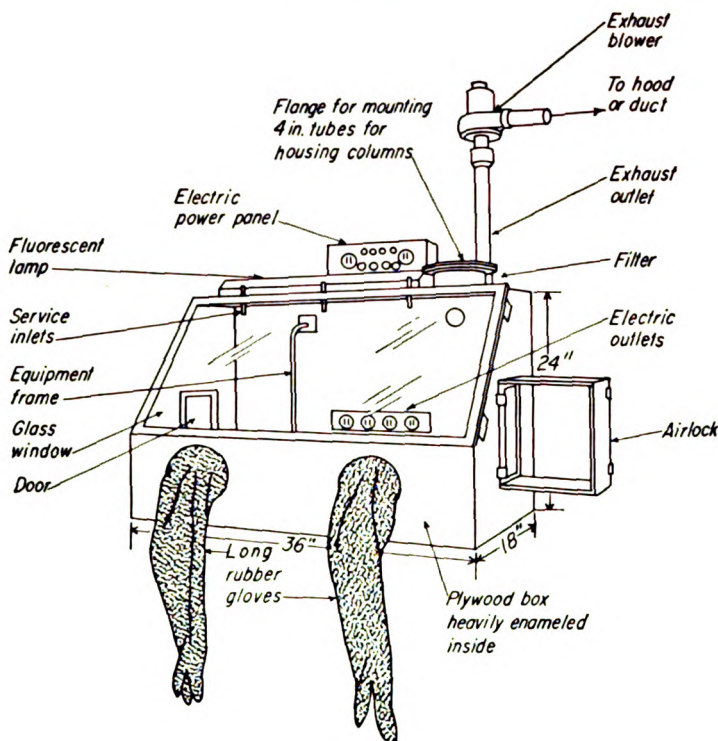


FIG. 9-3. Glove box.

VENTILATION

For a typical well-operated radioisotope laboratory, special ventilating systems are not required. However, care should be taken to ensure that incoming air is supplied in a manner that will prevent strong drafts or eddies. For programs involving very high levels of activity or significant levels of very hazardous materials, special **ventilating systems** are required. It is preferable for the air to flow from the areas of no activity to those in which higher levels are used. Therefore, exhaust demand should generally be slightly greater than supply volume, in order to create the desired air-flow pattern. Air subject to contamination should not be recirculated. Exhaust outlets should not be located near open windows or air intakes. In determining heating and air-conditioning requirements the added loads from the air discharged from the hoods should not be overlooked.

PLUMBING

The radioisotope program should be planned so that disposal by sewer is minimized. This will usually permit the disposal of the low concentrations of wastes through ordinary **plumbing** facilities. However, to ensure compliance with Federal and other regulations, special provisions may be required for collecting and storing

liquid wastes for controlled release to the sewers or for reconcentration. In laboratories where very high levels of activity or especially hazardous materials are handled it is often desirable to have emergency showers at strategic locations. In such cases the desirability of floor drains should be considered. Such a drain may be helpful in cleaning up spills but also might result in excess sewage contamination.

LABORATORY LAYOUT

Radioisotope laboratories often require considerably more **floor space** per worker than the conventional laboratory work. A requirement of 200 square feet per man is often recommended, but this figure may vary considerably. As previously indicated, added space is required for segregating work on basis of type and level of activity and nature of operation.

Storage requirements will vary widely depending upon the program. Enough shelf and cabinet space should be provided to ensure segregated storage of glassware and other apparatus used in the radioactive work to prevent its use in nonradioactive work. Specific areas should be designed or designated for this purpose. Where a variety of radioisotopes or high levels of activity are involved, a special storage room should be provided. For simplified programs, however, it may be adequate to set aside a portion of the high-level area for this purpose. For gamma-ray emitters a barricade shield should be constructed. Individual sources can then be kept within this barricade with or without individual shielding, depending upon protection requirements.

Some area should be designated as a **change area** for hanging smocks and for storage of gloves, respirators, or other specialized clothing and apparatus as needed. For the typical program, an area in the low-level room can be used for this purpose. However, for more involved programs, it is usually desirable to have a special area set aside for such purposes and for monitoring for contamination. If contamination potentials are so high as to require the use of coveralls, provisions should be made for a change room and for segregated storage of work and street clothes.

In designing a laboratory, attention should be given to the expected **traffic pattern**, particularly when there are several rooms. It is both annoying and sometimes hazardous to find that the hot area or the counting room has become a thoroughfare. An outside door to the storage area may be desirable for receiving isotope shipments, but unless traffic can be effectively controlled it is better to tolerate the slight inconvenience of bringing shipments in through the laboratory rather than to risk having this door used for primary access to the laboratory. It is usually better to limit access to the hot laboratory and/or storage area to a single door opening into a lower-level laboratory or change room rather than to an uncontrolled area such as a corridor or the outside of the building. Similarly it is sometimes better to have no direct access from the counting room to the laboratories. Although this may cause more traffic in the corridors, it usually reduces traffic in both the counting room and the laboratory.

The basic plan for the layout of a typical **radio chemical laboratory** is illustrated in Fig. 9-4. Work is segregated according to function and level of activity. This same plan is followed whether the work is carried out in a single room or in a number of rooms. In general, it is desirable to have separate rooms for working with dif-

ferent levels of activity. However, the room or area requirements depend upon the difference of the levels of activity, type of operation, and technical requirements. For example, in a radiochemical program, a one-room laboratory may be adequate

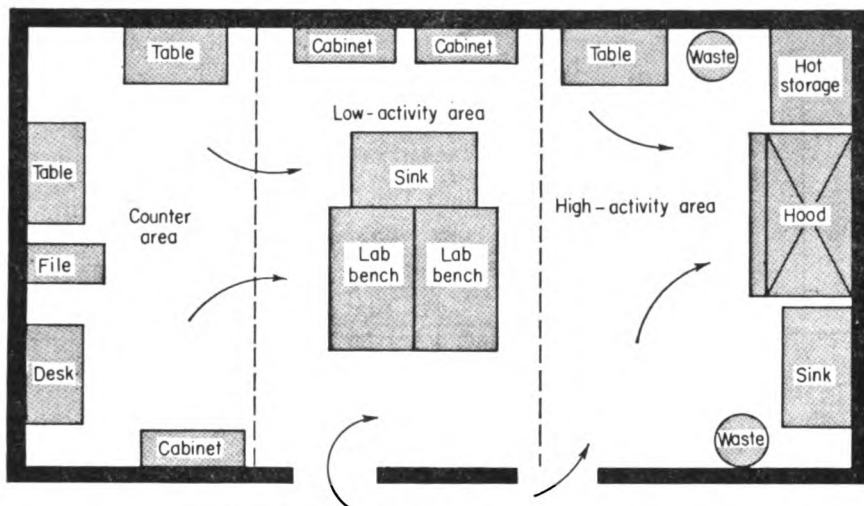


FIG. 9-4. Simple radiochemical laboratory.

to handle up to 100 millicuries of weak beta-particle emitters and up to a few millicuries of energetic beta-particle emitters or microcurie levels of gamma-ray emitters. The control of contamination and interference with counting procedures is simplified by adding a counting room (Fig. 9-5). If still higher levels of activity

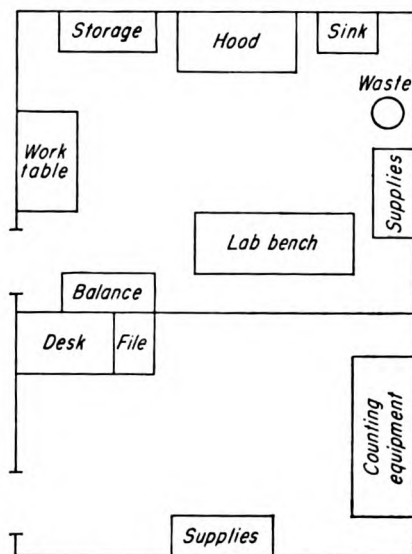


FIG. 9-5. Two-room radiochemical laboratory.

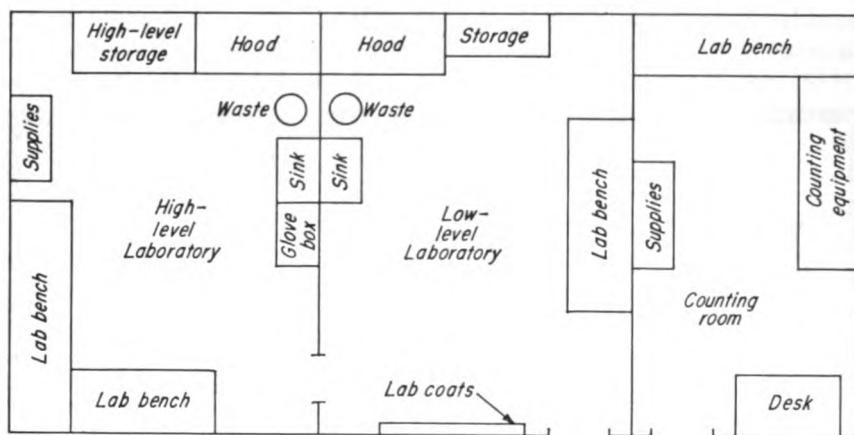


FIG. 9-6. High-level radiochemical laboratory.

are used, a third room should be provided so that the work can be further segregated (Fig. 9-6).

For **radiobiological laboratories** a room may be required for injection of the radioisotopes which have been prepared for administration to animals or application to plants, and still another low-activity room may be required for dissecting of animals and plants and preparing samples for analysis. A third room is desirable for counting (Fig. 9-7). It may be desirable to add a separate room for animals or

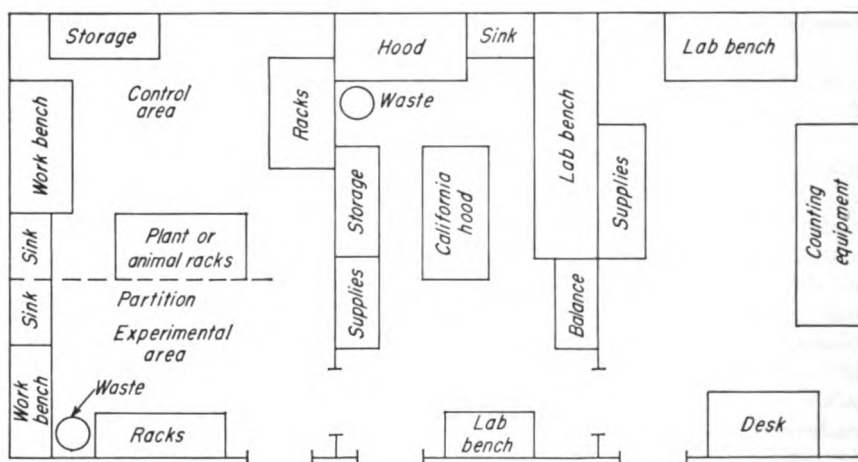


FIG. 9-7. Radiobiological laboratory.

plants, a radioisotope-storage room, a utility room for decontamination of cages or washing large apparatus, preparing seed beds, etc. For autoradiography, a dark-room should be added. In addition to these rooms such other rooms may be re-

quired as necessary for the housing of nonradioactive specimens, for chemicals, and for general-storage purposes.

For the **medical radioisotope laboratory** two rooms are generally desirable when materials are to be processed, calibrated, and prepared for administration. The first room is used for processing and storing radioisotopes. The second room is used for radioassay in standardizing doses, uptake measurements, and analyzing specimens (Fig. 9-8). This can be increased to three rooms if it is desired to make radioassays in a separate room from uptake measurements. On the other hand, the laboratory may be reduced to one room if the radioisotopes are obtained in prepared and precalibrated doses ready for administration to the patient.

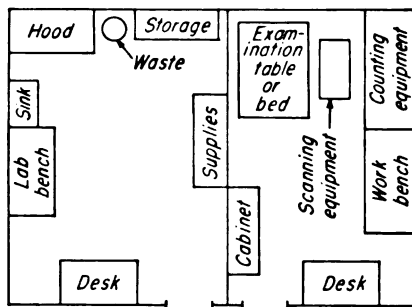


FIG. 9-8. Medical radioisotope laboratory.

For a more ambitious program, including therapy or research, several more rooms may be needed, depending upon the program. Such additional rooms may be used for receiving and storage, high-level dose preparation, low-level dose preparation, scanning patients, isolation of patients, autopsies, animal cages, production of labeled compounds, counting samples, etc.

The same principle of segregation of levels of activity and of general segregation of specialized procedures is desirable regardless of the type of laboratory. In all cases, special attention should be given to the technical requirements for preventing interference with measurements and counting procedures.

General References

Ward, Donald R., Design of Laboratories for Safe Use of Radioisotopes, *AEC Rept. AECU-2226*.

Health Physics Insurance Seminar, *AEC Rept. TID-388*.

"Laboratory Design for Handling Radioactive Material," Building Research Advisory Board, 1952.

"Safe Handling of Radioactive Materials," NBS Handbook 42, GPO.

"Control and Removal of Radioactive Contamination in Laboratories," NBS Handbook 48, GPO.

Harris, W. B., Laboratories, *Heating and Ventilating*, November, 1953, p. 74.

Schulte, H. E., E. C. Hyatt, H. S. Jordan, and R. N. Mitchell, Evaluation of Laboratory Fume Hoods, *Ind. Hyg. Assoc. Quart.*, 1954.

Comar, C. L., "Radioisotopes in Biology and Agriculture," McGraw-Hill, 1955.

- Radiochemistry Laboratory Symposium, *Ind. Eng. Chem.*, February, 1949.
- Morgan, G. W., Facilities and Equipment for Isotopes Program, *Hospitals*, March, 1955.
- Webster, S. H., E. J. Liljegen, and C. C. Powell, Hood for Radioactive Work, *Nucleonics*, vol. 10, no. 4, pp. 65-66, 1952.
- Van Bruggen, J. T., Cage-hood Assembly for Small Animals, *Nucleonics*, vol. 10, no. 3, pp. 64-66, 1952.
- Sognaes, R. F., and J. H. Shaw, A Small Self-contained Laboratory for Radioisotope Studies in Experimental Animals, *Arch. Ind. Hyg.*, vol. 3, p. 316, 1951.
- Ingraham, S. C., and W. R. Taylor, Radioisotope Facilities in the General Hospital, *Hospitals*, December, 1952, pp. 26-74.
- Wittrup, R. D., and R. L. Johnson, A Nursing Unit for Patients Receiving Radioactive Isotopes, *Hospitals*, vol. 28, pp. 65-88, May, 1954.
- Warren, S., and C. K. Spaulding, An Effective Isotope Suite, *Hospitals*, vol. 28, pp. 62-64, July, 1954.
- Hawkins, M. B., The Design of Laboratories for the Safe Handling of Radioisotopes, in "Laboratory Design," pp. 214-227, Reinhold, 1951.
- Symposium on the Handling of Radioactive and Toxic Substances, *Atomic Energy Research Establ. (G.B.) Rept. AERE C/R 958*.
- Cook, G. B., and F. Morgan, A Dry Box for Alpha, Beta, Gamma Work, *Atomic Energy Research Establ. (G.B.) Rept. AERE C/R 724*.

Section 10

RADIATION DETECTION AND MEASUREMENTS

By

HARALD H. ROSSI, Ph.D., *Radiological Research Laboratory, Columbia University, New York, N.Y. (Ionization Chambers)*

GEORGE A. MORTON, Ph.D., *RCA Laboratories, Princeton, N.J. (Scintillation Counters)*

SERGE A. KORFF, Ph.D., *New York University, New York, N.Y. (Proportional and Geiger Counters)*

GEORGE M. CORNEY, *Eastman Kodak Company, Rochester, N.Y. (Photographic Monitoring of Radiation)*

SANFORD C. SIGOLOFF, *Senior Scientific Executive, Edgerton, Germeshausen and Grier, Inc., Las Vegas, Nev. (Chemical Monitoring of Radiation)*

JAMES H. SCHULMAN, Ph.D., *U.S. Naval Research Laboratory, Washington, D.C. (Dosimetry Using the Optical Properties of Solids)*

Ionization Chambers.....	10-3
General Considerations.....	10-3
Ion Production and Ion Collection.....	10-4
General Design Features.....	10-6
Measurement of Charge or Current.....	10-7
Practical Examples.....	10-9
Special Types.....	10-11
Common Causes of Malfunction.....	10-13
Scintillation Counters.....	10-14
The Scintillation Counter.....	10-14
Principles of the Scintillation Counter.....	10-14
Scintillators.....	10-15
Photomultiplier Operation.....	10-21
Commercial Photomultipliers.....	10-24
Circuitry.....	10-26
Instruments and Applications.....	10-30

Proportional and Geiger Counters.....	10-33
Proportional Counters.....	10-33
Geiger Counters.....	10-41
Photographic Monitoring of Radiation.....	10-53
Photographic Considerations.....	10-53
Monitoring of X- and Gamma Rays.....	10-62
Beta-ray Monitoring.....	10-69
Neutron Monitoring.....	10-70
Monitoring of Mixed Radiations.....	10-73
Photographic Films for X-, Gamma-, and Beta-ray Monitoring.....	10-75
Chemical Monitoring of Radiation.....	10-76
Dosimetric Considerations.....	10-76
Evaluation Techniques.....	10-76
State of the Art.....	10-78
Dosimetry Using the Optical Properties of Solids.....	10-79
General Theory of Operation.....	10-79
Type I Dosimeters.....	10-81
Type II Dosimeters.....	10-83
General.....	10-84

IONIZATION CHAMBERS

Harald H. Rossi

GENERAL CONSIDERATIONS

The **ionization chamber** is the oldest instrument of radiation physics and still perhaps the most extensively used. Its main advantages are simplicity and wide range of sensitivity.

Ionization chambers have been constructed from a large number of materials and in a great variety of types, but basically an ionization chamber consists of at least two electrodes defining a collecting volume of gas. Any incident ionizing radiation will produce ion pairs in the gas. When a voltage difference is established, ions are drawn to the electrodes, and the resulting current or charge can be measured with appropriate apparatus.

The two basic functions of ionization chambers are detection and dosimetry.

Radiation detection, i.e., the indication of presence of radiation, usually involves no particular criteria beyond sensitivity. The latter may be enhanced by the simple expedient of making the chamber (or more precisely, the ionizable volume) large. Other methods of increasing the sensitivity include increase of gas pressure, decrease of electrical capacity, and appropriate choice of wall materials and gas filling.

When used for purposes of radiation hygiene, the function of the ionization chamber is **radiation dosimetry** rather than detection since it is commonly used to evaluate radiation fields in terms of exposure dose (roentgens) or absorbed dose (rads). If the radiation involved is ionizing per se (i.e., alpha or beta radiation) this imposes little limitation in the choice of wall materials. All that is required is a sufficiently thin window to admit the radiation. (In most designs this window forms at least part of one of the electrodes.) On the other hand, if the radiation produces ionization by secondary processes (e.g., X- and gamma rays or neutrons) the response of the chamber depends critically on the materials employed in construction. A chamber bounded by lead electrodes and filled with krypton will be much more sensitive to low- and medium-energy X-rays than a similar chamber with plastic electrodes that is filled with air. However, in the former case the response in terms of divisions per roentgen will be a strong function of X-ray energy while in the latter it will be largely independent of the kind of X-ray source involved.

Except for attenuation and multiple-scattering phenomena, the response of an ionization chamber depends on the nature of the gas and the characteristics of the wall up to a depth that is equal to the maximum range of any secondary ionizing particles that may be ejected from it. It is thus evident that in the case of dose

measurement these materials must respond in the same way or at least in a similar way as air or tissue, depending on whether roentgens or rads are to be measured. The degree to which air or tissue equivalence can be attained depends on the radiation involved. Thus, gamma rays of a few Mev are to be measured in terms of roentgens; the choice is not too critical. In the case of lower energies the problem of selection of materials is difficult. While the free-air chamber (see below) will yield a correct answer, only a judicious choice of solid materials may be expected to result in a chamber that is reasonably wavelength-independent.

A fundamental theorem that applies to ionization chambers is the **Bragg-Gray theorem**. This relates the energy imparted per unit mass of wall E_M to the ionization per unit mass of gas J_G by

$$E_M = J_G \rho_M W$$

where ρ_M is the relative-mass stopping power of wall and gas and W the average energy expended in the production of an ion pair in the gas (see below). In this general form the theorem is applicable only when the cavity is small compared with the particle range. When wall and gas are of identical atomic composition (homogeneous ionization chamber) this restriction does not apply and the above relation is simplified to

$$E_M = J_G W$$

Tissue-equivalent chambers are ionization chambers that contain in both wall and gas significant elements in the same concentration as they occur in tissue. Using the above relations, such chambers may be used for a more or less direct determination of absorbed dose due to any ionizing radiation.

Any dose measurement is of practical value only if the radiation field is substantially uniform throughout the dimensions of an ionization chamber. In certain instances radiation fields can be quite nonuniform, and employment of conventional chambers yields a meaningless average. This occurs particularly when one is dealing with radiation of quite limited penetrations, when the dose is to be determined near interfaces of differing density and/or atomic composition, and if the chamber is used in the immediate vicinity of a source. Thus, commercial survey instruments, although *sensitive* to beta radiation, are unsuitable for *dose* measurement. Depending on geometry and beta-ray energy, the skin dose may be underestimated by factors of 10 and more.

Ionization chambers usually measure a free-air or "first-collision" dose in that scattering and absorption are negligible. In the human body these factors are important and may lead to considerable variations. Thus, backscattering of soft X-rays may increase the skin dose by 50 per cent over the air dose while the dose received near the body axis is very much less. Determination of *tissue dose* or true absorbed dose must be carried out with mock-ups (*phantoms*) employing small ionization chambers inserted into tissue-equivalent material of appropriate shape.

ION PRODUCTION AND ION COLLECTION

The primary process in **ionization**, i.e., ejection of electrons, is much the same in all gases. The energy loss per unit distance traversed by a charged particle is proportional to the stopping power and density of the gas. The ratio of energy loss to

ionization is known as the *average energy expended in the production of an ion pair* W . In the case of air, the value is about 34 ev per ion pair for electrons and about 36 ev per ion pair for heavy particles (protons, alphas, etc.) For most gases of practical interest, the value of W does not differ greatly from the one for air, the variation being about ± 25 per cent.

In some gases the ejected electron will attach itself readily to neutral molecules to form negative ions. The most important practical examples are oxygen and water vapor. If such *electronegative gases* are present in minute concentrations in gases that do not exhibit electron attachment (such as nitrogen, the noble gases, and most hydrocarbons) electrons will still be trapped by the impurities. The mobility of positive and negative ions is of about the same order of magnitude (roughly 1 cm/sec per volt/cm), while the mobility of electrons is about one hundred times greater. Thus, in the presence of electronegative gases the electric field in the ionization chamber causes positive and negative ions to move comparatively slowly in opposite directions. If electronegative gases are absent, the positive ions will travel at about the same rate but the electrons will be swept out of the gas volume much more rapidly. It will be seen that in the latter situation recombination is less likely.

There are two types of **recombination**. One, known as *columnar recombination*, occurs among the ions generated along the trajectory of an individual charged particle. This process takes place before diffusion and the applied electric field have separated the ions sufficiently. It is of importance in the case of densely ionizing particles (such as protons or alphas). In the case of electrons, it is appreciable only at comparatively high gas pressures. Columnar recombination is independent of dose rate.

The second kind of recombination, *general (or intercolumnar) recombination*, takes place when ions drifting across the chamber volume encounter ions of opposite sign that have arisen along different particle tracks. The probability for this process is essentially proportional to the instantaneous concentration of ions in the chamber gas, which in turn is proportional to the dose rate. It is substantially independent of the specific ionization of the primary charged particles that traverse the chamber.

Recombination is lessened as the chamber voltage is increased, giving rise to the **saturation curve** (Fig. 10-1). From the foregoing, it is evident that the shape of this curve depends on a great number of factors (such as type of gas, type of charged particle, dose rate, and the variation of electric-field intensity inside the chamber). Usually the saturation curve exhibits a more or less distinct **plateau**, where the current is essentially independent of applied voltage. It is commonly assumed that when an ionization chamber is operated in this region all the ions liberated in the gas volume are collected. Actually there must always be a certain amount of recombination, although the fraction of ions lost may be inconsequential.

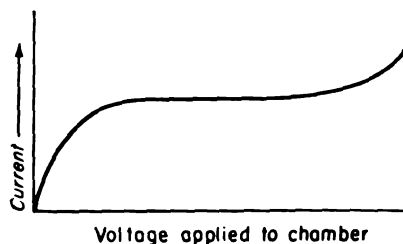


FIG. 10-1. Saturation curve.

At higher voltages **gas multiplication** sets in, i.e., electrons can acquire sufficient energy between collisions to ionize neutral gas molecules. Ionization chambers are usually not operated at such voltages or below saturation, although either region has been employed in chambers constructed for special purposes.

The primary ionizing particles usually have a preferential direction. Thus, gamma rays will eject electrons primarily in a forward direction. Since these electrons are charged, and since their energy is so great that they are virtually unaffected by the electric field in the chamber, they represent an irreversible current that can add to or subtract from the ionization current. In practice it will be found that virtually every ionization chamber exhibits a detectable **polarity effect**. When particle emission is strongly directional, and when the ionizable volume is small, the effect can become comparable with, and sometimes even larger than, the ionization current. In any precision measurement an ionization chamber must be operated at both polarities. The arithmetic mean of the two currents observed should be taken as the true ionization current.

Some sources emit ionizing radiation in bursts that may be short and intense. Thus the *instantaneous* dose rate near a betatron is about 5,000 times higher than the *mean* dose rate. It is evident that under such conditions saturation is much more difficult to attain. Special efforts must be made to ensure that at least most of the ions are collected. One of the most effective approaches is to utilize parallel-plate chambers having a narrow spacing between electrodes.

GENERAL DESIGN FEATURES

Most ionization chambers have either two or three electrodes. The two-electrode type (Figs. 10-3a and 10-4a) may be considered to consist of an accelerator and a collector, and the applied voltage appears across the insulation between these two members. In the three-electrode, or **guard-ring**, type (Figs. 10-3b and 10-4b) the collecting voltage appears between an accelerator and the (usually grounded) guard ring. The collecting electrode remains essentially at ground potential. This slightly more complicated construction results in at least two insulators. One, across the accelerating voltage, needs only to be good enough so that the battery drain is negligible. The other insulator must have low leakage, but this condition is more readily fulfilled since little or no voltage appears across it.

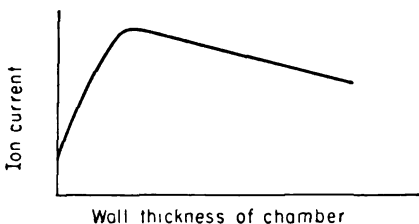


FIG. 10-2. Radiation equilibrium: transition curve.

For most purposes of dosimetry, ionization chambers are operated at **radiation equilibrium** (Fig. 10-2). If a beam of gamma radiation strikes the chamber wall, electrons are ejected and the number reaching the chamber interior increases until the wall is thick enough to stop the most energetic electrons.

Beyond that point, addition of further wall material eliminates more electrons than are produced because the primary gamma radiation is also attenuated. As the latter is much more penetrating, the curve is quite asymmetric. The optimum

wall thickness of an ionization chamber utilized for the measurement of electromagnetic radiation is therefore a function of energy. It will be noted that the curve in Fig. 10-2 does not start at the origin. This is meant to indicate that even at zero wall thickness a certain amount of ionization occurs because electrons ejected from the source or in the intervening air accompany most photon beams.

Analogous considerations apply to radiation equilibrium established by neutrons.

A great variety of materials have been used in the construction of chamber walls. These include gases (see free-air chamber, below), liquids, and solids. In most instances electric conduction is required, and unless the materials exhibit metallic or ionic conduction, thin coats of conducting substances (particularly colloidal carbon) are applied. As mentioned above, the choice of wall material is critical when ionization chambers are used in dosimetry of radiations that ionize by secondary processes. In the case of measurements in terms of roentgens (X- and gamma rays) one wishes to utilize an "air-wall" chamber. This is literally accomplished by the use of the free-air chamber, or effectively approximated by the use of such materials as light plastics, cardboard, or elektron metal (an alloy having the same average atomic number as air). In the case of measurements in terms of absorbed dose (all ionizing radiations), tissue-equivalent solids are employed. Tissue equivalence is particularly critical in the case of neutrons, and special gels or plastics must be employed.

Of the great variety of gases that have been utilized to fill the chamber interior, two mixtures are of particular importance in dosimetry: air for measurement of exposure dose, and tissue-equivalent gas for the measurement of absorbed dose in tissue. The latter may be obtained in a variety of mixtures that result in the desired relative concentration of hydrogen, oxygen, carbon, and nitrogen (the most important elements in soft tissue).

A variety of **insulators** have been employed. Some of these (such as sulfur) are only of historical interest. Amber and quartz have high resistivity, but the former is difficult to obtain and the latter unsatisfactory at high humidity. Ceresine, another material that has been employed for a long time, is satisfactory in most respects except for lack of mechanical strength. Most modern designs employ plastics such as polystyrene, polyethylene, teflon, Kel-F, or lucite. Most insulators exhibit polarization effects and also leak charge when exposed to ionizing radiation. Polystyrene seems least objectionable on this count.

To facilitate saturation, the electric field in the chamber should be as uniform as possible. A parallel-plate chamber approaches this requirement very well; however, cylindrical designs are often more practical. In this case the central electrode should have a rather large diameter, and sharp points and corners should be avoided. In addition, it is desirable to terminate the outer cylinder at both ends with curved (e.g., hemispherical) rather than plane surfaces. This tends to eliminate inside corners where the field is low.

MEASUREMENT OF CHARGE OR CURRENT

Depending on whether the detector employed in conjunction with an ionization chamber measures charge or current, it will register the total ionization that has occurred in a given time or the rate of ionization at any instant. The two pro-

cedures are, of course, equivalent when performed in a radiation field that is constant in time. On the other hand, a pocket dosimeter will register the total dose received as the wearer is exposed to a variety of conditions; this is the time integral of the dose rate as measured, for instance, by a survey meter. Examples of typical arrangements are given below.

Dose Meters. Charge is most conveniently measured in terms of the voltage change of a capacitor. Hence, dose meters are usually operated in conjunction with an electrometer (Fig. 10-3).

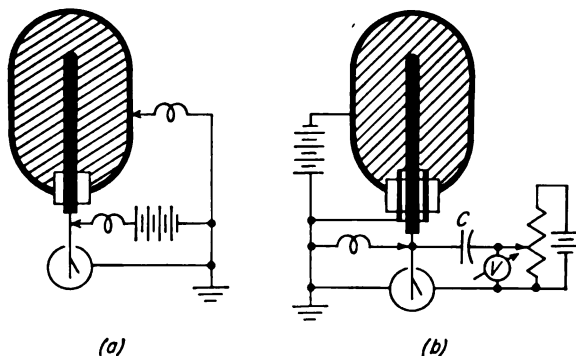


FIG. 10-3. Dose meters. (a) Condenser chamber. (b) Guard-ring chamber (Townsend method).

If the chamber has two-electrode construction, the applied voltage must be large enough to ensure saturation (usually of the order of 100 volts), and the voltage change is most readily determined if it is of the same order of magnitude,* although it must of course be less to maintain saturation conditions near the lower end of the scale. Since the voltage change is thus comparatively large, the capacity is kept small. For this reason the electrometer is often incorporated into the ionization chamber. However, for purposes of simplicity, the chamber is often detached when exposed to radiation. Figure 10-3a shows a typical arrangement in which this is possible. Chambers used in this manner are usually called condenser chambers. Electrometers employed are usually electrostatic devices (such as the quartz-fiber electrometer).

When a guard ring is employed the voltage rise detected is usually small since a large voltage between collector and guard ring would defeat the purpose of the latter. Thus, electrometers designed to measure small voltage differences (such as the Lindemann electrometer) are employed. Precision measurements often employ the electrometer as a zero indicator as shown in Fig. 10-3b, where the Townsend scheme is illustrated. The charge collected is balanced by opposite charge applied through a condenser C . When the potentiometer is adjusted until the electrometer shows zero deflection, the charge collected by the chamber is CV . It is to be noted that the capacity of chamber and electrometer does not influence the result except that it determines the sensitivity of the measurement.

* Schemes have been devised where this is not the case, but they are not commonly utilized.

Dose-rate Meters. Current-measuring devices frequently utilized are galvanometers or electronic electrometers.

Figure 10-4a shows a two-electrode chamber in conjunction with a galvanometer.

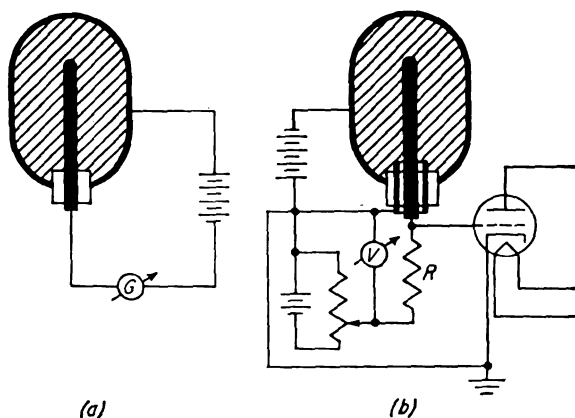


FIG. 10-4. Dose-rate meters. (a) Two-electrode chamber with galvanometer. (b) Guard-ring chamber and slide-back electrometer.

Figure 10-4b shows a guard-ring chamber with an electronic detector operated as null indicator. When the potentiometer is adjusted for zero deflection, the ionization current is V/R .

PRACTICAL EXAMPLES

The instruments described below are typical of a number that are readily available. Selection of any particular commercially available product should not be construed as endorsement in preference to other similar instruments.

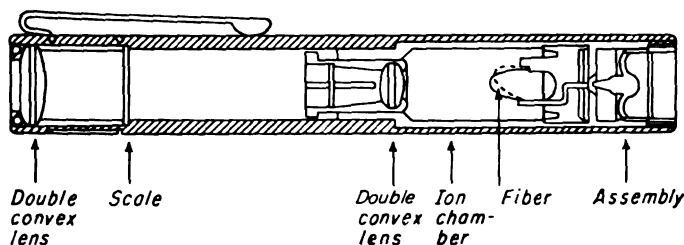


FIG. 10-5. Self-reading pocket dosimeter (fiber type).

Pocket Dosimeter. This device, worn for protection purposes, can be either self-reading or not. In the latter case it merely consists of a simple two-electrode chamber that is charged and read on a separate instrument. In the former instance a simple rugged quartz-fiber electrometer is mounted inside the ionizable volume and the fiber is observed through a small built-in microscope that has a scale in the eyepiece (Fig. 10-5). Since the leakage must be very low, a flexible diaphragm or

magnetic switch is incorporated into the instrument. The externally accessible insulation needs to be only good enough to maintain insulation during the brief period required for charging of the instrument. Typical full-scale sensitivities are 100 mr to 50 r.

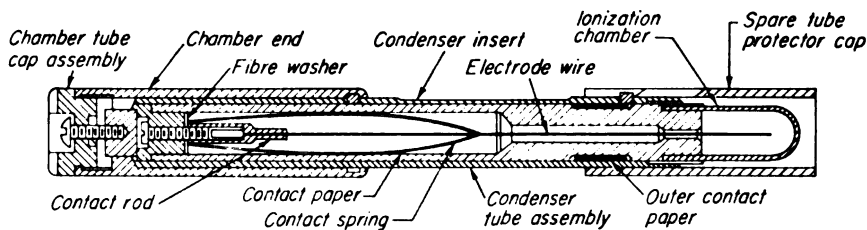


FIG. 10-6. Condenser thimble chamber. (Courtesy of Victoreen Instrument Co.)

Condenser Dosimeter. Used primarily for medical and biological dosimetry, this instrument consists of a charger-reader and one or more ionization chambers (Fig. 10-6). The chambers are detachable and of the condenser type. They range in full-scale sensitivity from a few roentgens to about 250 r. Chambers of still greater range often give trouble because of radiation effects on the dielectric and "parasitic" volumes. The latter are air volumes (usually between the chamber and its cap) where unwanted ionization can cause additional—and usually erratic—discharge of the chamber.

The charger-reader contains a quartz-fiber electrometer and either a frictional charger or a source of a d-c potential of 400 to 500 volts.

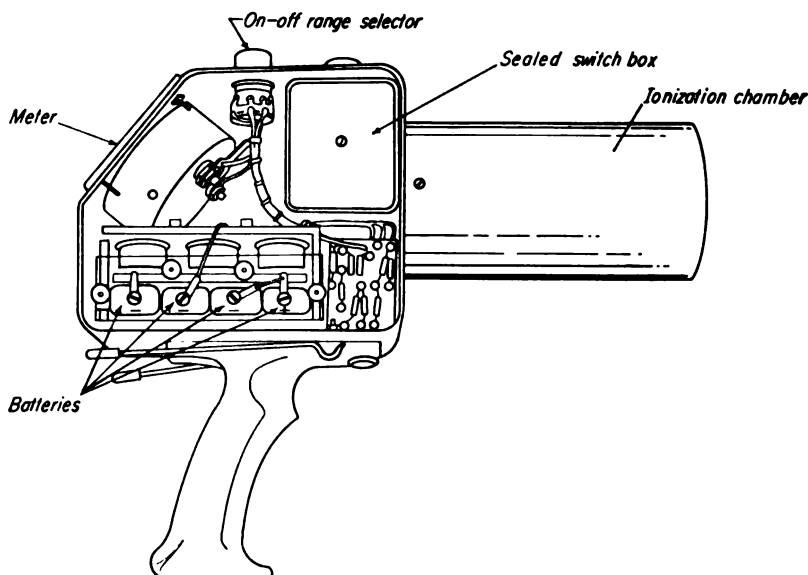


FIG. 10-7. Cutie Pie. (Courtesy of Tracerlab, Inc.)

Portable Survey Meter. The particular model shown in Fig. 10-7 is known as "Cutie Pie." It is a battery-operated instrument, incorporating a guard-ring ionization chamber and the simple two-stage d-c amplifier shown in Fig. 10-8.

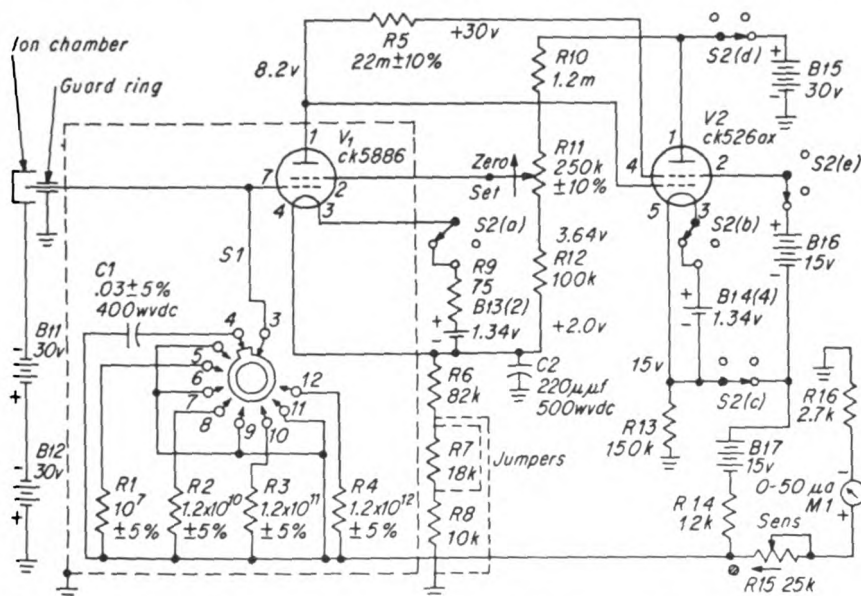


FIG. 10-8. Schematic diagram of Cutie Pie circuit. The elements within the broken lines are inside the sealed switch box. (Courtesy of Tracerlab, Inc.)

Sliding windows may be used to discriminate between alpha, beta, and gamma radiation, but for reasons mentioned above, the device should be used only for detection rather than for dosimetry of alpha and beta radiation.

SPECIAL TYPES

Two types of chambers will be described that are of particular importance in radiological physics.

Standard Free-air Chamber (Fig. 10-9). This is the primary standard for X-ray dose measurement. The beam is defined by a diaphragm and is permitted to pass through the space between a parallel-plate ionization chamber with no solid materials intervening. The lower plate consists of a collecting section that is insulated from two flanking guard plates. The latter serve to define an accurately cylindrical collecting volume (usually of rectangular cross sections). Employment of parallel wires at appropriate potentials reduces "fringing" of the electrical field still further.

The number of ions collected is equal to the one generated by electrons that arise from the crosshatched volume. While some electrons may leave the collecting volume, they are compensated for by others that enter it. The distances from the diaphragm to the edge of the collecting volume and the distance from the edge of

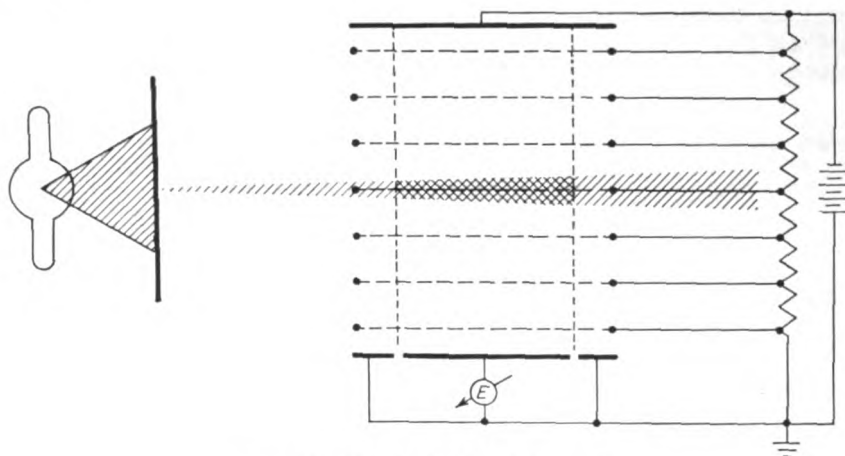


FIG. 10-9. Standard free-air chamber.

the beam to the nearest plate must be more than the maximum electron range. Under these conditions the total collected charge in esu divided by the doubly cross-hatched volume is equal to the dose in roentgens (see Sec. 2) after pressure and temperature corrections are applied.

Extrapolation Chamber. This instrument, devised by Failla, is used to obtain an accurate value of dose in a radiation field that varies rapidly in one direction. There are a number of practical examples of this situation, such as the skin dose due to X-rays or the surface dose near extended sources of beta radiation. The device is a parallel-plate chamber with variable spacing.

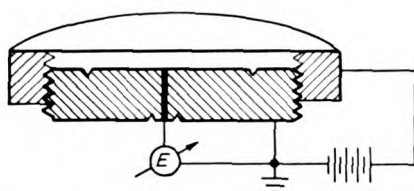


FIG. 10-10. Failla extrapolation chamber (half section).

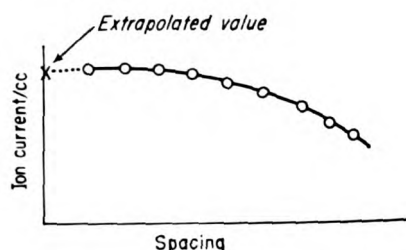


FIG. 10-11. Extrapolation curve.

The embodiment shown in Fig. 10-10 consists of a thin accelerating electrode provided with an insulator ring having an internal thread. The lower disk provided with an external thread is made of insulating material, but painted with colloidal graphite. A thin scratch divides the top surface into a collector and a guard ring; a pencil lead establishes electrical connection between the collector and the electrometer.

The ionization current is measured as a function of electrode spacing, the latter being varied by relative rotation of the two members. If the current per unit

volume is plotted as a function of electrode spacing, a graph as shown in Fig. 10-11 is obtained, and it is usually not difficult to extrapolate the curve obtained to zero spacing.

COMMON CAUSES OF MALFUNCTION

The ionization chamber is a comparatively rugged device, and it will often be found that improper performance is due to deficiency of associated equipment or circuits rather than due to any shortcomings of the chamber proper.

The most common difficulty arising in chamber operation is failure of the insulation. Insulators must often attain resistance of the order of 10^{18} ohms to keep leakage at acceptable levels. While such resistance can be readily obtained with absolutely clean surfaces, the slightest amount of foreign matter—such as a simple fingerprint—can render an insulating surface ineffective. Often it will be found that barely visible threads or hairs span, and effectively short-circuit, the insulator surface. While most insulators when properly cleaned can operate satisfactorily at 100 per cent humidity, they appear to lose this quality with time by mere exposure to the atmosphere. Presumably dust or perhaps oily deposits are responsible. If the insulator has been cleaned with the aid of a detergent and inadequately rinsed, it is particularly sensitive to humidity. In addition, a dried soap film can develop an emf; hence insulators may exhibit a current without applied voltage. When an insulator is cracked it usually must be discarded since dirt and moisture cannot be removed. Most insulators are piezoelectric, and flexure or other strains (sometimes caused by thermal expansion) will produce spurious currents.

The collecting system of an ionization chamber is highly sensitive to electric pickup and must be adequately shielded during measurement. In addition, variations in the interelectrode capacitance (such as vibration of electrodes) or of the voltage applied across it (such as an unsteady power supply) will induce charges on the collector.

Poor conductivity of electrodes may cause sluggish or erratic response. Conducting coatings can peel off and sometimes even fall across the insulation.

"Parasitic volumes" occur whenever the collecting system is exposed to gas spaces other than the chamber volume, and if such spaces are traversed by radiation. The unwanted ion current is often also erratic since the electric field in parasitic volumes is likely to be far from saturation.

SCINTILLATION COUNTERS

George A. Morton

THE SCINTILLATION COUNTER

In nuclear research the scintillation counter is one of the most important devices for the detection and measurement of radiation. Many laboratories working in this field use it in large numbers because it possesses certain performance characteristics which cannot be equaled by any other instrument. As yet, this detector has not been applied in the field of radiation hygiene to nearly so great an extent. In the past the scintillation counter has not been a standard commercial product, and it has been necessary to build individual laboratory models as they were needed. Recently, more and more of these instruments have been appearing on the market in the form of well-engineered standardized units. As this trend continues and accelerates, it is certain that the scintillation counter will become recognized as an instrument of major importance in the domain of radiation hygiene.

The three attributes of this form of detector which make it so very important are:

Sensitivity. It is the most sensitive detector for almost all forms of nuclear radiation. It is particularly well suited for the detection of gamma rays. Over a large part of the gamma-ray spectrum, a well-designed instrument can have a counting efficiency of nearly 100 per cent.

Energy Discrimination. It is capable of indicating the energy of nuclear particles or photons. An estimate of the resolution which can be obtained with a spectrometer (employing an NaI:Tl scintillator) over the gamma- or beta-ray energy range from a few tenths of a Mev to several Mev can be made with the aid of the equation

$$R = \frac{6.5}{\sqrt{E}}$$

where E is the energy in Mev and R the percentage width of the energy-distribution curve at half maximum.

Time Resolution. It has the best time-resolution capabilities of any instrument known. Time dispersion of less than a millimicrosecond is practical.

PRINCIPLES OF THE SCINTILLATION COUNTER

The scintillation counter shown schematically in Fig. 10-12 consists of a transparent scintillator, usually in the form of a phosphor, optically coupled to a photo-

multiplier which in turn is coupled to appropriate output circuitry.* The phosphor may be in the form of a single crystal, a block of plastic, or a liquid. When a particle or photon of nuclear radiation enters the volume of the scintillator, it gives up its energy, producing excitation which generates a scintillation, or flash of light. The visible-light photons thus produced strike the photocathode of the multiplier, causing the release of photoelectrons. The photoelectrons enter the structure of the photomultiplier, where their number is increased by a very large factor as a result of the secondary-emission processes occurring at the cascaded stages. An anode

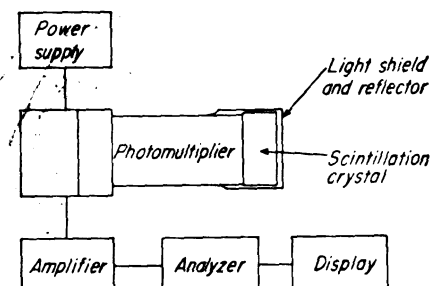


FIG. 10-12. Schematic diagram of scintillation counter.

collects the amplified electron current, and it is brought out of the multiplier through an appropriate lead. The output of the photomultiplier is supplied to associated presentation circuitry which analyzes the electrical signal and displays the results.

SCINTILLATORS

The **excitation** of a scintillator by the energy of nuclear particles usually involves high-speed electrons moving through the material. The exact mechanism depends upon the type of nuclear radiation.

For *beta rays*, which are high-energy electrons themselves, each electron is scattered as it moves through the material, producing intermediate-velocity electrons which, in turn, excite the visible-light-producing transitions.

Gamma rays interact with the scintillator as though they were discrete particles (photons). The relationship between the number of photons, the photon energy, and the amount of radiation in roentgens is shown in Fig. 10-13. The interaction may occur in three different ways.† The first is **Compton collisions** wherein the gamma-ray photons interact with relatively free electrons, giving some of their energy to the electrons (thus exciting the phosphor) and continuing as lower-energy

* Kallmann, H., *Natur u. Tech.*, July, 1947.

Coltman, J. W., and F. H. Marshall, A Photomultiplier Radiation Detector, *Phys. Rev.*, vol. 72, no. 6, p. 528, 1947.

Birks, J. B., "Scintillation Counters," McGraw-Hill, New York, 1953.

Morton, G. A., Recent Developments in the Scintillation Counter Field, *Proc. Intern. Conf. Peaceful Uses Atomic Energy, Geneva*, vol. 14, August, 1955. (See also *Trans. PGNS*, IRE, 1956.)

† Segré, E. (ed.), "Experimental Nuclear Physics," Wiley, New York, 1953.

scattered photons. Unless the volume of the scintillator is large enough to absorb the scattered photon, only part of the primary energy is given up to the scintillator. The dependence of Compton scattering upon the atomic number Z of the phosphor, the mass A of the atom, and the density ρ is given by

$$\tau_c \sim \frac{\rho Z}{A}$$

The second type of interaction is the so-called **photoelectric absorption**. Here the gamma ray interacts with a relatively firmly bound electron. In this case, the

atom to which the electron is bound can absorb some of the momentum of the photon, permitting the electron to receive all the energy of the photon without violating the law of conservation of momentum. In this type of absorption process, the entire photon energy goes to exciting the scintillator. The dependence of this type of absorption upon material is given by

$$\tau_{ph} \sim \frac{\rho Z^m}{A} \quad 4 < m < 4.5$$

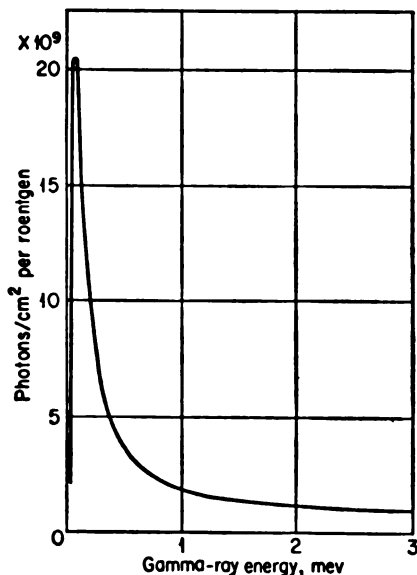


FIG. 10-13. Photon-radiation intensity relationship.

The third type of interaction is **pair production**. In this interaction, the gamma-ray photon interacts with matter in such a way as to produce a pair of oppositely charged particles, an electron and a positron. The creation of the pair uses about 1 Mev of energy of the initial particle. The rest of the energy appears as kinetic energy of the created particles. All this kinetic energy goes into exciting the phosphor. The

positron annihilation releases two rather low-energy gamma-ray photons which in a scintillator of reasonable volume and density will be absorbed and contribute to the excitation. Therefore, the excitation of the phosphor corresponds to the energy of the incident photon. The law governing this interaction is given by

$$\tau_{pair} \sim \frac{\rho Z^2}{A}$$

Figure 10-14 shows the variation of absorption with photon energy for these three processes in carbon (approximating that for any organic compound) and in lead (characteristic of a very heavy inorganic scintillator).

For **alpha particles**, because of their low penetrating power, the scintillator usually takes the form of a rather thin layer (Graves, J. D., L. A. Webb, and R. H. Davis, Performance of Lucite-bonded Alpha Scintillation Screens, *Nucleonics*, vol. 10, no. 12, p. 68, 1952). The layer thickness is chosen so as to just

absorb all the energy of the alpha particle since any greater thickness would simply increase the optical loss in the material and, even more serious, would make the screen more sensitive to a gamma-ray background.

For **high-energy neutrons**, a hydrogenous organic phosphor will be fairly efficient (Emmerich, W. S., A Fast Neutron Scintillator, *Rev. Sci. Instr.*, vol. 25, pp. 1, 68, 1955). Unfortunately, such a material is also an efficient gamma-ray scintillator which makes a background discrimination difficult. The neutron excitation occurs as a result of the motion of protons through the material where collisions between the neutrons and hydrogen nuclei have occurred. Such accelerated protons are frequently referred to as "knocked-on" atoms, and as they move through the lattice giving up their energy, some of it goes into electronic excitation which results in fluorescence.

For low-energy neutrons, such as thermal neutrons, it is necessary to include in the phosphor an element which has a large capture cross section for slow neutrons and which undergoes a nuclear reaction as a result of capturing a neutron.* Lithium, boron, cadmium, and indium are elements having isotopes which exhibit this property.

Where **speed and good time resolution** are major considerations in the scintillation counter, the decay time of the phosphor is of paramount importance. This, for example, is the case when coincidence techniques are used to separate certain types of wanted nuclear phenomena from a background of unwanted nuclear events. The fast phosphors are generally organic materials and exhibit an exponential decay characteristic in their fluorescence. The rate of production of optical photons is given by

$$\frac{dN}{dt} = \frac{N_0}{\alpha} e^{-t/\alpha}$$

and the number of optical photons N emitted after time t by

$$N = N_0(1 - e^{-t/\alpha})$$

Both the decay constant α and the total number of emitted photons N_0 appear in the equations describing the resolving time of a scintillation counter.

* Schenck, J., Neutron Detecting Phosphors, *Nucleonics*, vol. 10, no. 8, p. 54, 1952.

Muehlhause, C. O., Neutron Scintillation Counters, *Nucleonics*, vol. 14, no. 4, p. 38, 1956.

Sun, K. H., P. R. Malmberg, and F. A. Pecjak, High Efficiency Slow Neutron Scintillation Counter, *Nucleonics*, vol. 14, no. 7, p. 46, 1956.

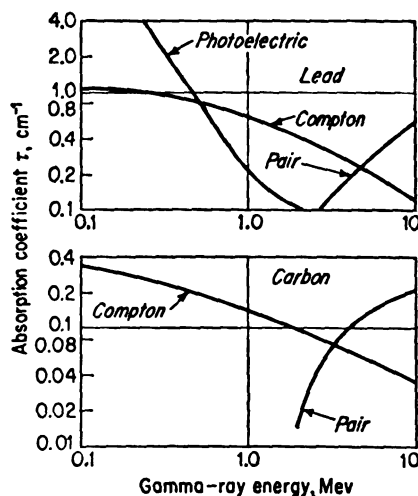


FIG. 10-14. Types of gamma-ray absorption.

In **energy spectrometry** for the determination of gamma-ray photon energy, in general a crystal is used which has as high a density as practical in order to maximize photoelectric conversion.* Sodium iodide activated with thallium is excellent for this purpose. Furthermore, the volume of the scintillator is made large enough to make total absorption of the photon energy probable. Frequently the scintillator will have a deep well or cup cut in it into which the specimen may be placed so that essentially all the gamma rays emitted by the specimen are absorbed.

There are two main classes of **practical scintillators**, namely, inorganic scintillators and organic scintillators.†

The action of most **inorganic scintillators** involves the excitation of electrons into the conduction band of a semiconductor (or into exciton states) and the random motion of the excited carriers through the material until they are captured by a radiative transition into an activator center. This process of fluorescence can be very efficient but is, in general, rather slow so that the material has a very long time constant.

The most widely used inorganic phosphor is thallium-activated **sodium iodide**.‡ It is prepared in the form of a completely transparent single crystal and can now be obtained commercially in very large sizes (4- to 6-in. crystals are not infrequent, and crystals have been made up to 10 in. in diameter). The high density of the material and the high atomic number of the iodine give it a high absorption coefficient to gamma rays. Some of these same characteristics result in a very favorable ratio of photoelectric to Compton absorption of radiation. This, together with its very high fluorescent efficiency, makes it one of the best materials known for scintillation-counter spectrometry. The material is very sensitive to moisture, and a crystal of sodium iodide will be irreparably damaged if exposed to air of high humidity even for a relatively short time. This means that sodium iodide crystals used for scintillation counters must be hermetically sealed in containers whose walls are transparent to the radiation which the device is to detect and must have one or more transparent windows to allow the optical photons of the scintillations to reach the phototube. Usually, the material is sealed or potted in a light aluminum can whose walls are coated with white magnesium or titanium oxide powder to have high diffuse reflection and into one end of which is sealed a glass window. Techniques for encasing crystals have been developed which are very satisfactory and give the crystals long working life.

The fluorescence from **organic scintillators** is usually the result of electron excitation within individual molecules.§ The fluorescent process, while less efficient than

* Borkowski, C. J., and R. L. Clark, Gamma Ray Energy Resolution with NaI:Tl Scintillation Spectrometers, *Rev. Sci. Instr.*, vol. 24, no. 11, p. 1046, 1953.

† Swank, R. K., Characteristics of Scintillators, *Ann. Rev. Nuclear Sci.*, vol. 4, p. 111, 1954.

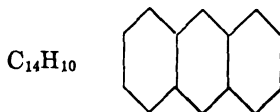
Birks, J. B., "Scintillation Counters," McGraw-Hill, New York, 1953.

‡ Hofstadter, R., Alkali Halide Scintillation Counters, *Phys. Rev.*, vol. 74, no. 1, p. 100, 1948; see also *Phys. Rev.*, vol. 75, no. 5, p. 796, 1949, and Van Sciver, W., Spectrum and Decay of NaI, *Nucleonics*, vol. 14, no. 4, p. 50, 1956.

§ Swank, R. K., Recent Advances in Theory of Scintillation Phosphors, *Nucleonics*, vol. 12, no. 3, p. 14, 1954.

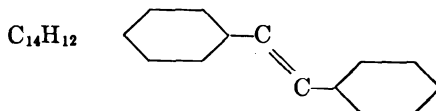
that of sodium iodide, is very much faster. Organic phosphors are used in the form of single crystals, liquid scintillators, and plastic scintillators.

Anthracene, with the following structure,



in the form of a single crystal is very frequently used for scintillation counting.* The material can be readily grown into large and clear crystals. In order to obtain efficient scintillators, it is necessary to use highly purified anthracene and rather involved techniques are used to obtain the requisite purity. The material is quite versatile since, while it has only about half the efficiency of sodium iodide, its time constant is smaller by nearly a factor of 10. It can therefore be used in problems where a combination of pulse-height discrimination and good time resolution is required.

Trans-stilbene is another organic phosphor which is frequently used in the form of a single crystal. It has the following chemical composition:



Like anthracene, it can be grown into large, clear single crystals without too much difficulty, although it does require rigorous purification and the growing must be carried out in such a way as not to permit oxidation. The fluorescent efficiency of trans-stilbene is about one-third that of anthracene, but its speed is very much higher. The decay constant for trans-stilbene at room temperature is about 6×10^{-9} sec. The time constant decreases if the material is cooled. This phosphor is very valuable where good time resolution is essential as in the case of accurate coincidence measurements.

Liquid scintillators are an important class of organic phosphors.† Here, an organic solvent such as xylene or toluene has dissolved in it a small amount of activator. These activators are actually organic phosphors such as terphenyl, diphenyloxazole, and tetraphenylbutadiene. The nuclear particle or photon releases energy in the solvent. This energy is rapidly transferred through the volume of the scintillator and is captured by the activator centers to be released in the form of optical photons. The high efficiency of this energy-transfer mechanism is indicated by the fact that a 1 per cent solution of terphenyl in xylene has nearly the fluorescent efficiency to gamma rays of a single crystal of terphenyl. In general, however, liquid scintillators are not as efficient as the best single-crystal organic

* Robinson, W. H., and W. Jentschke, Response of Anthracene Scintillation Crystals to Monoenergetic Soft X-rays, *Phys. Rev.*, vol. 95, no. 6, p. 1413, 1954.

† Kallmann, H., and M. Furst, Fluorescence of Solutions Bombarded with High Energy Radiation, *Phys. Rev.*, vol. 79, no. 5, p. 857, 1950; see also *Phys. Rev.*, vol. 81, no. 5, p. 853, 1951; vol. 85, no. 5, p. 816, 1952; vol. 97, no. 3, p. 583, 1955.

scintillators. They are, however, very fast, and decay constants of the order of 2 or 3 msec can readily be obtained. By adding a third organic material such as diphenylhexatriene or diphenylstilbene, frequently known as wavelength shifters, the efficiency of a liquid scintillator may be increased by as much as 50 per cent. A wavelength shifter, however, slightly increases the time constant of the phosphor.

In addition to their speed, the ease with which large-volume scintillators can be made and the flexibility of their geometry are important factors in the value of liquid phosphors.* Very large liquid scintillators are frequently used in radiobiology for the purpose of completely immersing experimental animals, etc., thus permitting very accurate counting of the total content of radioactive atoms in the specimen.

Plastic phosphors (scintillators) are very similar to the liquid scintillators.† They can be made in large sizes and in any shape and have the advantage of retaining their shape without having to be enclosed in a liquid-tight container as does a liquid scintillator. Their speed and efficiency are about the same as those of liquid scintillators.

Table 10-1 lists some of the frequently employed inorganic and organic scintillators together with a number of their performance characteristics.

Table 10-1. Scintillator Characteristics

Scintillator	Density	Index of refraction	Spectral maximum, Å	Time const, μ sec	Relative efficiency
Sodium iodide (NaI:Tl).....	3.67	1.7745	4,100	250	100
Cadmium tungstate.....	7.90	2.2-2.3	5,200	10 ³	(100?)
Anthracene.....	1.25	1.59	4,400	36	48
Trans-stilbene.....	1.16	1.622	4,100	6	28
Xylene + terphenyl					
+ diphenylhexatriene.....	0.86	1.500	4,500	5	23
Xylene + terphenyl.....	0.86	1.500	4,000	3	16
Polyvinyltoluene + terphenyl					
+ diphenylstilbene.....			4,400	5	23
Polystyrene + tetraphenylbutadiene.....	1.06	1.595	4,000	3	17

* Pringle, R. W., and Turchinets, Liquid Scintillator Techniques for Radiocarbon Dating, *Rev. Sci. Instr.*, vol. 26, no. 9, p. 859, 1955.

† Buck, W. L., and R. K. Swank, Efficient Plastic Scintillators, *Nucleonics*, vol. 11, no. 11, p. 48, 1953.

Eichholz, G. G., Preparation of Plastic Scintillation Phosphors, *Rev. Sci. Instr.*, vol. 23, no. 1, p. 34, 1952.

Fischer, J., Preparation of Large-area Plastic Scintillators, *Nucleonics*, vol. 13, no. 5, p. 52, 1955.

PHOTOMULTIPLIER OPERATION

The operation of a photomultiplier can best be described with the aid of the simplified schematic diagram shown in Fig. 10-15. The photocathode P is usually a semi-transparent layer of material, capable of emitting electrons under the influence of optical photons, on the inside of the glass end of the tube. The emitted electrons are accelerated away from the cathode and focused onto the first secondary-

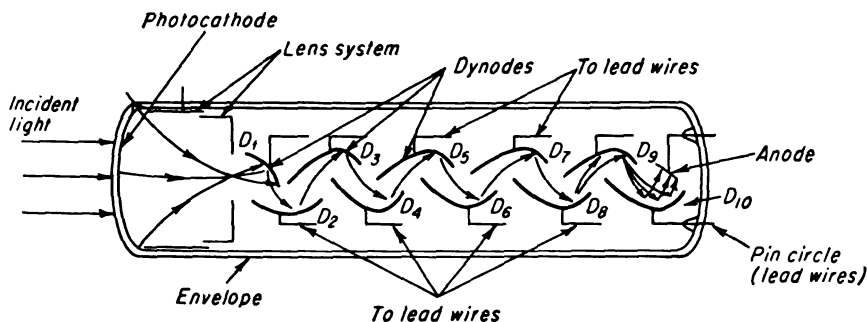


FIG. 10-15. Schematic diagram of photomultiplier.

emitting electrode or dynode D_1 by means of the electron optical system consisting of one or more cylindrical electrodes or grids maintained at different positive potentials. These potentials generally are between that of the cathode and that of first dynode. The active area of the first dynode is coated with a material capable of emitting several secondary electrons for each incident electron. The dynodes D_1, D_2, D_3 , etc., are so shaped that electrons emitted by D_1 are focused onto D_2 , those emitted by D_2 are focused onto D_3 , etc. This process is repeated for the k dynodes making up the multiplier structure. The electrons emitted by the last dynode are collected on anode A and are brought out of the tube in the form of a current through lead O . If a dynode produces an average of σ electrons as the result of the bombardment by each electron, it is said to have a secondary-emission ratio σ . The secondary-emission ratio depends upon the nature of the coating on the dynode and the energy of the bombarding primary electrons. Figure 10-16 shows the variation of σ with primary electron voltage for cesium-antimonide and cesium-activated silver magnesium, both of which materials are widely used as the sensitizing agent for dynodes in commercial photomultipliers.

It is obvious from the description of the multiplier operation that the ratio of the output current I_0 to the primary photocurrent I_p will be given by

$$\frac{I_0}{I_p} = \sigma^k = G = \text{gain}$$

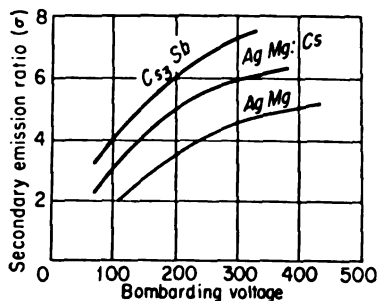


FIG. 10-16. Secondary emission ratio.

The primary photocurrent depends upon the intensity and spectral composition of the light incident upon the *photocathode* and the material used to sensitize it.* Figure 10-17 gives the spectral-response characteristic of several photoemitters used in photomultipliers. The sensitivity of the photocathode of a multiplier is usually expressed by two quantities, the number of microamperes per lumen from a

tungsten source operated at a color temperature of 2870°K, and the number of microamperes produced by light from a similar source which is filtered with a Corning No. 5113 filter of one-half stock thickness.

For **spectrometry** and other applications of the scintillation counter where the **pulse height** or amplitude of the output pulses is used as a measure of the energy of the incident particle, it is important to have the response of the multiplier as nearly linear as possible. The photoemission from the cathode is strictly proportional to the incident light up to large light values. The secondary-emission ratio is for all practical purposes independent of current. However, in the later stages of a multiplier, the density of the space current between dynodes and between the last dynode and collector may become quite high, giving rise to nonlinearity due to space charge. This is particularly true with the high-gain multipliers used where speed is of importance. It should be borne in mind that, when accurate pulse-height measurements are required, the peak instantaneous output current from a conventional photomultiplier should not exceed

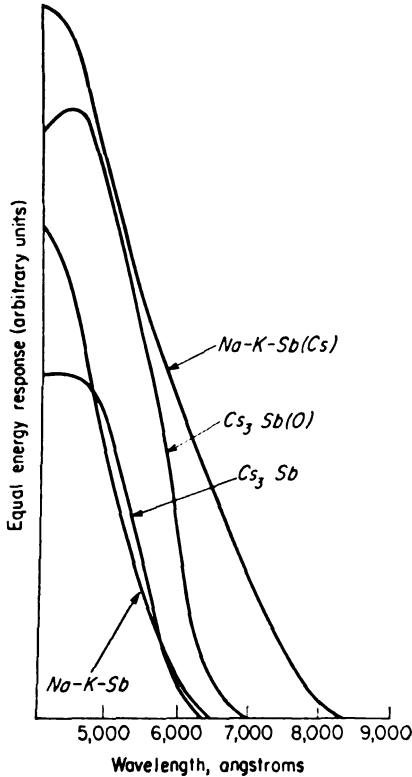


FIG. 10-17. Cathode spectral response.

a few milliamperes. The relationship between the percentage width $W_{1/2}$ of the pulse-amplitude distribution curve at its half maximum when N_0 photons strike the cathode and the multiplier characteristics is

$$W_{1/2} = 236 \left[\frac{1}{N_0 \gamma \eta_p} \left(1 + \frac{\epsilon}{\sigma - 1} \right) \right]^{1/2}$$

where γ = quantum efficiency of cathode

η_p = photoelectron collection efficiency

σ = secondary-emission ratio per dynode

ϵ = distribution coefficient ($\epsilon = 1.5$)

* Sommer, A. H., New Photoemissive Cathodes of High Sensitivity, *Rev. Sci. Instr.*, vol. 26, no. 7, p. 725, 1955.

The **time resolution** of a photomultiplier can be made very high.* The time resolution is determined not by the length of time it takes for the output pulse to appear after photoelectrons have been released from the cathode, but rather by the variation in this time from one pulse to the next. This transit-time spread is entirely a problem of electron ballistics, and it has been shown that secondary emission does not contribute to it. The transit-time spread between one dynode and the next varies from a rms value of 0.3 to 8 or 10 μsec depending upon the multiplier structure. The total time spread and the probable error in time resolution depend upon this interdynode time spread, the secondary emission of the dynodes, the number of dynodes, and the way in which the multiplier is operated. An approximate expression for the rms time $\langle t \rangle$ error of a scintillation counter is

$$\langle t \rangle = \sqrt{\frac{\alpha\beta}{N_e}} \left[\frac{4\sigma}{(\sigma - 1)^2} (2\sigma - 1) \right]^{1/4}$$

where α = decay constant of scintillator

β = interdynode transit-time spread of electrons in photomultiplier

N_e = number of photoelectrons released from cathode by scintillation

This time resolution is obtained when the trigger point of the timing circuit in terms of electrons from the cathode is set at n electrons where

$$n = N_e \frac{\beta}{\alpha} \left(\frac{\sigma}{2\sigma - 1} \right)^{1/4}$$

Values of $\langle t \rangle$ and n for three organic phosphors are given in Table 10-2.

Table 10-2

	N_0	α	n_{opt}	$\langle t \rangle$, μsec
Anthracene.....	500	36	2	0.3
Trans-stilbene..	300	6	5	0.15
Liquid.....	100	3	3	0.2

Assume $\sigma = 4$, $\beta = 0.2 \times 10^{-9}$ sec.

For the **detection of very small amounts of radiation**, one of the factors which limits a scintillation counter's sensitivity is the spurious pulses generated by the photomultiplier. These pulses may be due to thermionic emission from the cathode, ionization of residual gas in the tube, and cold discharge between electrodes. The last two mentioned sources of spurious pulses are not fundamental and have been almost wholly eliminated from modern commercial multipliers when they are operated in their normal voltage range. The **thermionic emission** from the cathode

* Morton, G. A., Time Resolution of Scintillation Counters, *Nucleonics*, vol. 10, no. 3, p. 39, 1952.

Post, R. F., Resolving Time of Scintillation Counters, *Nucleonics*, vol. 10, no. 6, p. 56, 1952.

is a fundamental property of photoemitters. It depends upon the type of cathode and its temperature. For an S11 (Cs_3Sb) photocathode, which is by far the most widely used photoemitter in commercial multipliers, the thermionic emission is about 5,000 electrons/sec/cm² at room temperature. It can be reduced by cooling the cathode, the rate of decrease being of the order of a factor of 2 for each 10°C of cooling. Unless especially prepared, most cathodes cannot be cooled below about 100°K without becoming so resistive as to interfere with the operation of the multiplier. Cooling to dry-ice temperature is, however, frequently employed for multipliers used to detect small amounts of soft radiation (e.g., carbon and tritium counting).

COMMERCIAL PHOTOMULTIPLIERS

The type of photomultiplier selected for use in a given scintillation counter depends largely upon the purpose for which the counter is being designed.*

For **general-survey-type instruments** or for a flexible general-purpose instrument the RCA 6199 photomultiplier is very useful. It can be efficiently coupled to a sodium iodide crystal 1 to 1½ in. in diameter and the combination mounted in an easily handled probe. Where a somewhat larger probe and crystal (e.g., a 2-in.-diameter crystal) are required, the Du Mont 6292 or the RCA 5819 (or 6655) is very satisfactory. For very large crystals (e.g., 3 and 5 in. diameters), the Du Mont 6363 (3 in.), the Du Mont 6364 (5 in.), or the RCA 7046 (5 in.) is useful. For very small probes, the RCA developmental tube C7204, which is ¾ in. in diameter, will find application.

For **energy spectrometry**, the Du Mont 6292 has been found to have the best pulse-height resolution capabilities. The RCA 6342 and 6655 are both good tubes for spectrometry and are nearly an order of magnitude better in time-resolution capabilities where the problem requires both speed and energy discrimination. For problems involving larger crystals, e.g., total-absorption spectrometers, the Du Mont 6363 and 6364 and the RCA 7046 are very useful tubes.

For situations where the maximum time resolution is required, the best tube appears to be the RCA 7264. This tube has a much higher gain than those mentioned above and, in general, obviates the need of a fast distributed amplifier. The large tube RCA 7046 is also very fast and has the same gain as the 7264. This tube should be used where large phosphors and speed are necessary, as may be the case for certain coincidence measurements.

Table 10-3 lists some of the principal commercial photomultipliers which are available in the United States and gives their more important operating characteristics.

* Much information can be obtained from the technical literature furnished by the manufacturers of photomultipliers.

Table 10-3. Characteristics (Approximate) of Representative Commercial Photomultipliers

Type	Max outside dimensions, in.		Cathode			Dynode system			Over-all voltage	Light equivalent of noise, lumens
			Area, cm ²	Response	Avg sensitivity, $\mu\text{a/l}$	No. of dynodes	Material	Gain		
	Diam	Length								
RCA:										
1P21	1 $\frac{1}{2}$ "	3 $\frac{1}{2}$ "	1.9	S11	40	9	Cs ₃ Sb	8.3×10^6	1,250	5×10^{-13}
6199	1 $\frac{1}{2}$ "	4 $\frac{1}{2}$ "	7.75	S11	45	10	Cs ₃ Sb	2.8×10^6	1,250	4×10^{-12}
6903	2 $\frac{1}{2}$ "	6 $\frac{1}{2}$ "	14.2	S13	60	10	Cs ₃ Sb	0.4×10^6	1,250	6.7×10^{-12}
5819	2 $\frac{1}{4}$ "	5 $\frac{1}{2}$ "	14.2	S11	50	10	Cs ₃ Sb	2.3×10^6	1,250	7×10^{-11}
6342	2 $\frac{1}{4}$ "	5 $\frac{1}{2}$ "	14.2	S11	60	10	AgMg	0.55×10^6	1,500	7×10^{-12}
6655	2 $\frac{1}{4}$ "	5 $\frac{1}{2}$ "	14.2	S11	50	10	Cs ₃ Sb	2.3×10^6	1,250	7×10^{-12}
6372	2 $\frac{1}{2}$ "	7 $\frac{1}{4}$ "	80	S11	33	10	Cs ₃ Sb	2.5×10^6	1,200	1×10^{-11}
7264	2 $\frac{3}{4}$ "	7 $\frac{1}{2}$ "	14.2	S11	60	14	AgMg	2×10^7	2,300	6×10^{-12}
7046	5 $\frac{1}{4}$ "	10 $\frac{1}{4}$ "	100	S11	50	14	AgMg	6×10^6	3,400	
RCA developmental tubes:										
C7204 *	3 $\frac{1}{4}$ "	3 $\frac{1}{4}$ "	1.6	S11	40	9	Cs ₃ Sb	2×10^5		
Du Mont										
6467	1 $\frac{1}{2}$ "	4 $\frac{1}{2}$ "	5	S11	60	10	AgMg	2×10^6	1,800	
6291	1 $\frac{1}{2}$ "	5	7.9	S11	60	10	AgMg	2×10^6	1,800	
6292	2 $\frac{1}{2}$ "	5 $\frac{1}{2}$ "	11.5	S11	60	10	AgMg	2×10^6	1,800	
6363	3 $\frac{3}{4}$ "	6 $\frac{1}{2}$ "	31.5	S11	60	10	AgMg	2×10^6	1,800	
6364	5 $\frac{1}{2}$ "	7 $\frac{1}{2}$ "	88	S11	60	10	AgMg	2×10^6	1,800	
7064	2 $\frac{1}{2}$ "	5 $\frac{1}{2}$ "	11.5	S11	60	10	CsSb	7.5×10^6	1,300	
7065	1 $\frac{1}{2}$ "	5 $\frac{1}{2}$ "	7.9	S11	60	10	CsSb	7.5×10^6	1,300	
Du Mont special multiplier phototubes:										
K1323	1,000	S11	40	12	AgMg	8×10^5 at 105 volts/stage			Large crystals	
K1209	88	S11	60	12	AgMg	2×10^6 at 95 volts/stage			For low-level detection	
K1213	37	S11	60	12	AgMg	2×10^6 at 95 volts/stage			For low-level detection	
K1295	11.5	S11	60	12	AgMg	2×10^6 at 95 volts/stage			For low-level detection	
K1306	Like 6292 but with S13 response									
K1323	11.5	S11	95	10	AgMg	2×10^6 at 145 volts/stage			Extra-sensitive photocathode	

* Approximate data.

CIRCUITRY

The circuitry associated with a multiplier can be divided into two parts.* The first is that pertaining to the **source of voltage** used to supply the dynodes and focusing electrodes. In general, this will consist of a rather well regulated voltage supply capable of delivering 1,000 to 3,000 volts depending upon the multiplier used. This voltage is divided into the required steps to supply the electron lens system in the vicinity of the cathode and the k dynodes of the multiplier.

The second part of the circuitry generally consists of an **amplifier**, the analyzing circuits, and the presentation means. The amplifier, which receives the output pulses from the multiplier, must be quite fast (i.e., time constants of $\frac{1}{2}$ to $2 \mu\text{sec}$ for detection and spectrometric work), should be linear, and should be capable of accepting a wide range of pulse amplitudes without blocking. Where the scintillation counter is a simple detector, the amplifier is usually followed by a pulse-height discriminator and either a pulse-rate measuring circuit or a pulse-counting circuit. For spectrometric work, the analyzing circuitry is a pulse-height analyzer feeding into one or more pulse-counting circuits. For problems involving short time resolution, the amplifier must be very fast (e.g., a distributed amplifier), and the analyzing circuitry generally takes the form of a coincidence circuit or a time sorter.

The three classes of output circuits outlined in the preceding paragraph by no means cover the wide variety of circuits used in scintillation counters and only serve to indicate those which are most widely used at present.

The principal requirement of the **power supply** is very good regulation. This need for high regulation arises from the fact that the gain of a multiplier varies with a high power of the over-all voltage. A 10-stage multiplier as normally operated will

vary with the seventh power of the voltage, and a 14-stage multiplier with the tenth power. Therefore, a 1 per cent change in over-all voltage will produce a 7 per cent change in gain in the former and a 10 per cent change in gain in the latter.

For battery-operated instruments, the voltage is obtained by interrupting the current from low-voltage batteries with a vibrator so that a transformer and rectifier can be used to generate the voltage

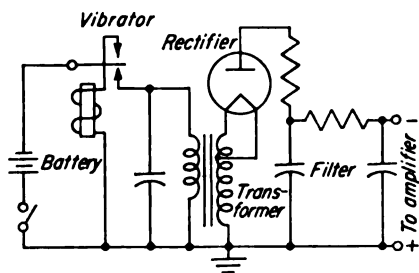


FIG. 10-18. Vibrator battery power supply.

needed for the photomultiplier. Where regulation is required, it is accomplished by means of a high-voltage glow-discharge regulator tube. A high-resistance bleeder divides the voltage into the steps needed to supply the multiplier. Figure 10-18 is a schematic diagram of such a power supply. Alternatively, the voltages may be supplied from high-voltage batteries.

* Scarrott, G. G., *Electronic Circuits for Nuclear Detectors*, *Progr. in Nuclear Phys.*, vol. 1, p. 73, 1950.

Elmore, W. C., *Electronics for the Nuclear Physicist*, *Nucleonics*, vol. 2, no. 2, p. 4; no. 3, p. 16; no. 4, p. 43; no. 5, p. 50, 1948.

Gas tubes, vacuum tubes, or transistor oscillators offer an alternative means for obtaining the pulsed or alternating current required to operate the step-up transformer.

Where the counter is to be operated from conventional 110-volt a-c service, the power supply may be of the usual type with a step-up transformer and rectifier capable of producing 20 or 25 per cent more voltage than is required for the multiplier so that an absorption regulator can be employed to supply well-regulated voltage to the multiplier. This is diagramed in Fig. 10-19.

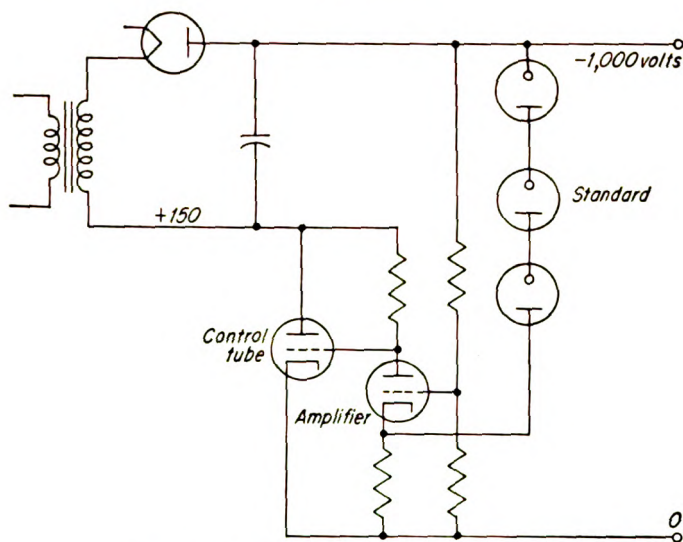


FIG. 10-19. A-C power supply and regulator.

Where possible, the bleeder or voltage divider used to obtain voltages for the multiplier dynodes should follow good conservative practice. It is better to waste power in a low-impedance bleeder than to risk incorrect voltage distribution due to electrical leakage in the multiplier and associated circuitry. For a multiplier whose output current is not to exceed 1 ma and which employs 100 to 200 volts per stage, it is quite common to use 100,000-ohm resistors between stages in the divider. If there is any reason to expect the occurrence of accidental momentary high current anywhere in the multiplier, it is frequently advisable to put high-valued resistors in series with each of the dynodes to limit the maximum current that can flow. Figure 10-20 shows a bleeder and protective resistances suitable for use with an RCA 5819 or 6342 photomultiplier.

When the multiplier involved is one employing sharp electrostatic focus (as is the case with all the RCA photomultipliers), it is possible to arrange the divider in such a way that the gain is independent of the over-all voltage over quite a wide range of voltage change.* Figure 10-21 shows a circuit arrangement for accomplish-

* Morton, G. A., Photomultipliers for Scintillation Counting, *RCA Rev.*, vol. 10, no. 4, p. 525, 1949.

ing this. The fixed voltage may be obtained from a battery or a gas-discharge regulator tube. Included with the figure is the curve showing gain as a function of

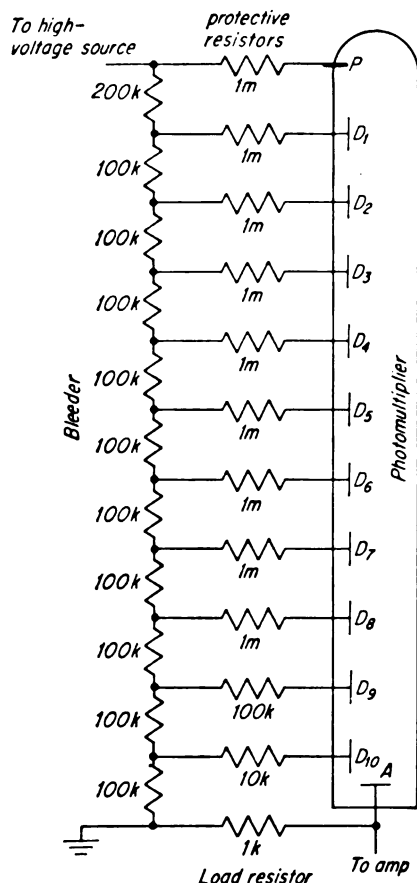


FIG. 10-20. Bleeder and protective resistances.

hundred or more channels, each selecting a different narrow range of pulse heights and supplying them to individual counter or memory circuits.† A well-engineered

over-all voltage. The voltage range where the gain is constant is marked on this diagram. Where the multiplier has a large number of stages as with the 6810, it may be necessary to apply compensating circuits to two sets of dynodes instead of one in order to obtain sufficient control.

Presentation Circuitry. For almost all applications of multipliers with 10 dynodes or less, it is necessary to amplify the output pulse before it can be used to actuate analyzing and counting circuits. The design of suitable pulse amplifiers is rather specialized, and anyone undertaking it should review the literature on the subject.* The conventional amplifier will have one or more preamplifier stages, possibly in the form of cascaded pairs followed by a linear pulse amplifier so designed that it will not block under overload and that its gain does not change with rate of pulsing.

For almost all applications, the amplifier is followed by some form of discriminator, a circuit designed to allow only pulses above a certain amplitude to pass, or a pulse-height selector which only allows pulses in a narrow range of heights to pass. The schematic diagram of a basic discriminator is shown in Fig. 10-22. Frequently, for energy spectrometry, the analyzer may have a

* Chase, R. L., and W. A. Higinbotham, A Flexible Pulse Amplifier with Good Overload Properties, *Rev. Sci. Instr.*, vol. 23, no. 1, p. 34, 1952.

Fairstein, E., Nonblocking Double-line Linear Pulse Amplifier, *Rev. Sci. Instr.*, vol. 27, no. 7, p. 475, 1956.

† Francis, J. E., P. R. Bell, and J. C. Gundlach, Single Channel Analyzer, *Rev. Sci. Instr.*, vol. 22, no. 3, p. 133, 1951.

Kelley, G. G., Pulse Amplitude Analyzers for Spectrometry, *Nucleonics*, vol. 10, no. 4, p. 34, 1952.

Higinbotham, W. A., Today's Pulse-height Analyzers, *Nucleonics*, vol. 14, no. 4, p. 61, 1956.

multichannel pulse-amplitude analyzer with magnetic-core storage has the same order of complexity as a modern digital computer.

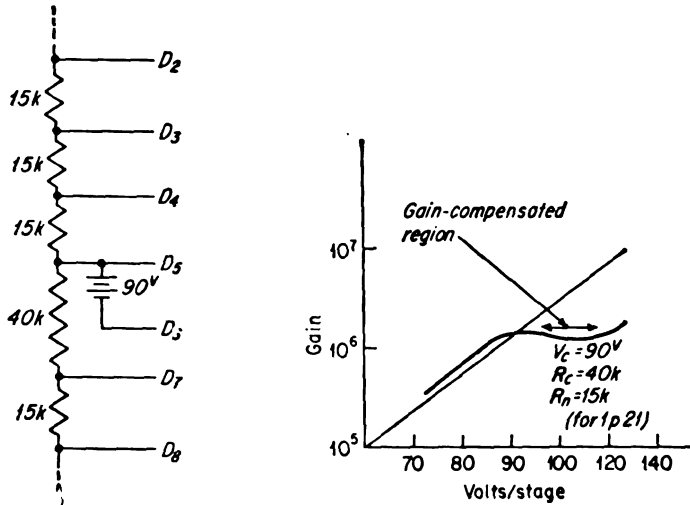


FIG. 10-21. Gain compensation.

Frequently, the time relationship between nuclear events is the physical quantity to be measured. Here, either fixed- or variable-delay coincidence circuits are employed. A coincidence circuit is a circuit arrangement which gives an output pulse only if pulses are supplied to input terminals simultaneously (i.e., separated by

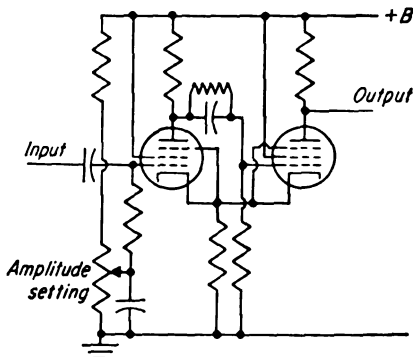


FIG. 10-22. Basic pulse-height discriminator.

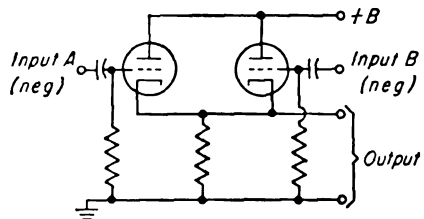


FIG. 10-23. Coincidence circuit.

less than a predetermined time interval).^{*} Figure 10-23 illustrates the elements of one type of frequently used coincidence circuit. If a time interval separates the two

^{*} Bell, R. E., Nuclear Particle Detection: Fast Electronics, *Ann. Rev. Nuclear Sci.*, vol. 4, p. 93, 1954.

nuclear events to be observed, a fixed- or variable-delay line may be placed between one multiplier and the coincidence circuit.*

The output from the analyzing circuits which have been described above is in the form of pulses where the desired information is either their rate of occurrence or their total number. The most commonly used pulse-rate indicator consists of a circuit element which converts the output pulses into current pulses of equal amplitude and duration and a means for measuring the average current from this element. Where the pulses are to be counted, the rate is usually quite high so that a very fast counter must be employed. The simplest form of counter is a sequence of binary pairs (i.e., vacuum-tube "flipflop" circuits) followed by a mechanical register. Other more complicated counting mechanisms may be required for pulse-amplitude analysis work such as a Williams storage tube, magnetic-core storage arrays, or mercury delay-line loops. Finally, a good oscilloscope often is all that is needed to display the information from a scintillation counter.

INSTRUMENTS AND APPLICATIONS

The variety of scintillation-counter instruments and the range of applications of this device is so large that only a few classes can be indicated here.† The basic scintillation-counter radiation detector because of its very high sensitivity will eventually be a very important instrument in the radiation-hygiene field. This instrument consists of a detector head with a sodium iodide crystal and photomultiplier which may be mounted in a separate probe or may be part of the instrument box, together with the associated circuitry which consists of a battery-operated power supply, an amplifier, and a rate meter.

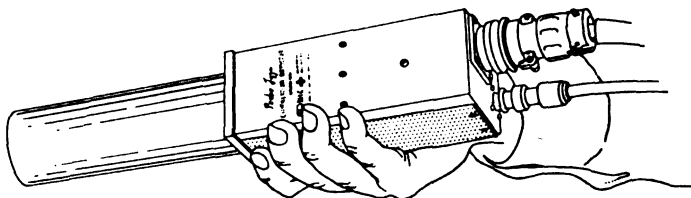


FIG. 10-24. Scintillation detector probe. (Courtesy of Baird-Atomic, Inc.)

Next to this in importance in the field are the scintillation counters for radioactive-tracer work.‡ Here, the scintillator is usually given a specific configuration to suit it to the analysis being undertaken. The output circuitry includes a pulse-height selector, set to correspond to the energy of the radiation from the tracer elements being sought so as to reduce background effects, and a counter rather

* Lundy, A., Delayed Coincidence Circuit for Scintillation Counters, *Rev. Sci. Instr.*, vol. 22, no. 5, p. 324, 1951.

† AEC Catalog (unclassified but not published for general distribution), "Radiation Measuring Instruments and Components," Radiation Instrument Branch, Division of Biology and Medicine, AEC, July, 1954.

‡ Kahn, B., and W. S. Lyon, Scintillation Spectrometer in Radio Chemical Analysis, *Nucleonics*, vol. 11, no. 11, p. 61, 1953.

than a rate meter. This represents the simplest type of instrument that might be used for tracer work. For complicated analysis, the use of an instrument employing a multichannel pulse-amplitude analyzer, which permits the identification of contributions from a number of different radioactive materials, may be required. Where the radioactive element being detected involves a low-energy gamma or beta ray such as is the case with carbon or tritium counting, very elaborate precautions may be required to reduce external and internal spurious counts.* Some of the precautions taken are the use of uncontaminated shields and "umbrellas" of scintillation counters or Geiger counters surrounding the detector which is actually making the analysis and placed in anticoincidence with it so that cosmic rays and environmental radioactivity will not be counted. Cooling the cathode of the photomultiplier or two multipliers in coincidence looking at the same crystal are often employed to reduce the effect of thermionic emission.

Coincidence instruments may also be required in radiation hygiene. One example is that of potassium used as a biological tracer in the location of brain tumors. Potassium is a positron emitter, and the annihilation radiation is in the form of two gamma-ray photons emitted in opposite directions. By using a pair of scintillation counters in coincidence, the position of the tracer can be accurately determined because it must lie on a line between the two counters when it is detected.

Scintillation counters are successfully used in the detection of alpha-emitting elements. These are difficult to detect because of the very short range in air and because alpha particles will penetrate only the thinnest of windows. The most efficient type of scintillation counter for detecting alpha particles consists of a multiplier with a large-area cathode and the scintillator in the form of a thin layer (5 mg/cm^2) of zinc sulfide powder bonded directly on the cathode window. If the instrument is to be used where there is appreciable ambient light, it is necessary to coat the zinc sulfide screen with a lighttight film. This film must be extremely thin

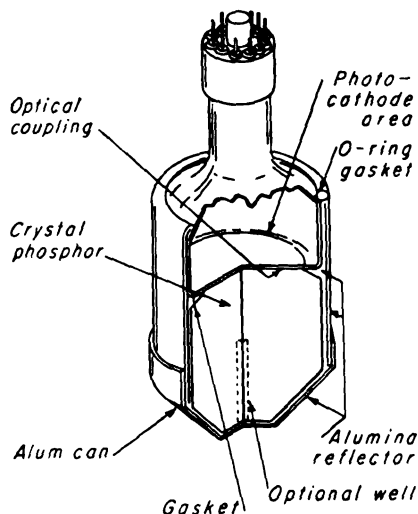


FIG. 10-25. Commercial photomultiplier with integrally mounted phosphor. (Courtesy of Levinthal Electronic Products, Inc.)

* Haigh, C. P., Gamma-ray Scintillation Counter for Weak Radioactive Solutions, *Nucleonics*, vol. 12, no. 1, p. 34, 1954.

Johnston, W. H., Low-level Counting Methods for Isotopic Tracers, *Science*, vol. 124, no. 3226, p. 801, 1956.

Hays, F. N., D. L. Williams, and Betty Rogers, Liquid Scintillation Counting of Natural C^{14} , *Phys. Rev.*, vol. 92, no. 2, p. 512, 1953.

Rosenthal, D. J., and H. O. Anger, Liquid Scintillation Counting of Tritium and C^{14} Labeled Compounds, *Rev. Sci. Instr.*, vol. 25, no. 7, p. 670, 1954.

to allow penetration of the alpha particles. A double layer of beaten aluminum leaf has been used as a lighttight protective coating for this class of instrument.

The instruments outlined above are only a few of the existing instruments which have application in the field of radiation hygiene. A great many other instruments

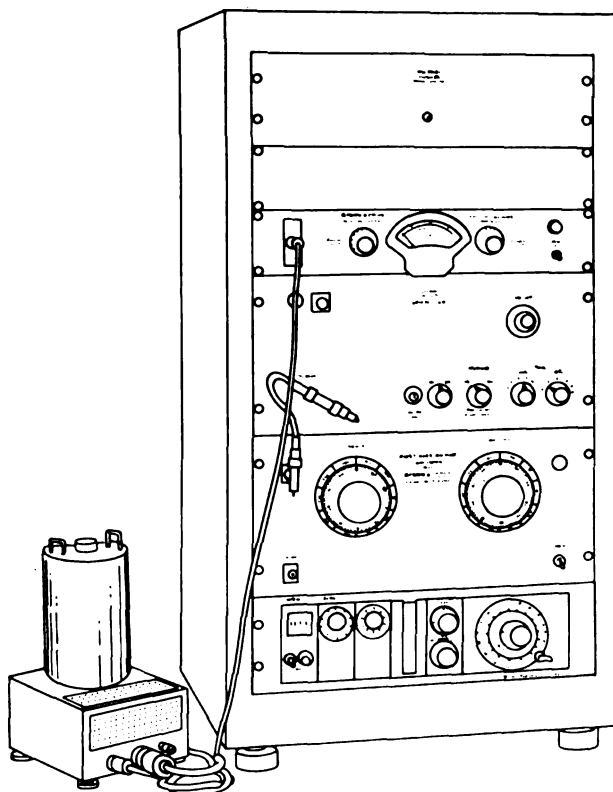


FIG. 10-26. Single-channel differential scintillation spectrometer. (Courtesy of Baird-Atomic, Inc.)

applicable to this field have already been devised, and their number is increasing rapidly as the merits of this type of device are becoming more widely known and as the components of which it is constituted are being improved, standardized, simplified, and made more reliable.

Examples of commercially available scintillation instruments are shown in Figs. 10-24, 10-25, and 10-26.

PROPORTIONAL AND GEIGER COUNTERS *

Serge A. Korff

General References

Korff, S. A., "Electron and Nuclear Counters," 2d ed., Van Nostrand, Princeton, N. J., 1955.

Rossi, B., and H. Staub, "Ionization Chambers and Counters," McGraw-Hill, New York, 1949.

Montgomery, C. G., and D. D. Montgomery, *Phys. Rev.*, vol. 57, p. 1030, 1940.

Trost, A., *Z. Physik*, vol. 105, p. 399, 1937.

Korff, S. A., and R. D. Present, *Phys. Rev.*, vol. 65, p. 274, 1944.

Steuer, H. G., *Phys. Rev.*, vol. 61, p. 38, 1942.

Maze, R., *J. phys. radium*, vol. 7, p. 164, 1946.

PROPORTIONAL COUNTERS

In this part of Sec. 10, **proportional counters** are discussed. Geiger counters are covered elsewhere. It is assumed that much of the pertinent matter, such as the processes of ion collection and the basic considerations in the operations of ionization chambers, is already familiar to the reader. Further information on the use of such instruments is to be found in other sections. This discussion, therefore, is limited to the actual counters themselves and to their characteristics of operation.

Proportional and Geiger counters were first used in about the year 1910. They cannot, therefore, be regarded as novel devices. However, with the great mass of new knowledge about gas-discharge phenomena, and with the modern development in electronic instrumentation, a great deal has been learned about these devices in recent years, and today we are passing from the stage in which the use of the device was an art to that in which it is a science (Korff, S. A., "Electron and Nuclear Counters," 2d ed., Van Nostrand, Princeton, N. J., 1955).

In the following, let us assume that we are dealing with an arrangement of electrodes, a cylinder with a wire stretched along its axis, between which two electrodes a potential has been applied. It is assumed that the wire is positive with respect to the cylinder and that the whole arrangement is placed in a suitable container, such that a known and controllable amount and kind of gas exists therein. It is further assumed that the wire is connected to a suitable electronic device such that pulses appearing upon the wire may be detected, their amplitude measured,

* Work was in part supported by ONR and AF contracts.

and the number of each size be recorded. As is well known, variations in the above conditions can occur and can be useful; however, such exceptions will be specifically so mentioned. A suitable window is provided, if necessary, such that the entity to be detected can enter the sensitive region and there will produce an ionizing event.

The discharge mechanism of a proportional counter is similar in many respects to the operation of an ionization chamber. As is known to the reader who is familiar with the operation of ionization chambers, once an ionizing event is produced in the sensitive volume, the electrons formed in this event will be impelled by the field to drift toward the central collecting electrode (the central wire), and there it will cause a change in potential of this electrode which change in potential we shall call a pulse. Now if the voltage across the counter is increased, the electrons will move faster, gaining more energy on their way to the wire. As we continue to increase the potential, the electrons will attain an energy sufficient to ionize the atoms with which they collide on their way in, and will in this way create more electrons. Each additional electron thus formed will also drift toward the central wire and will by collisions create still further electrons. Thus each original electron will create a **cascade** or **avalanche** of electrons. The magnitude of the pulse obtained on the central wire therefore depends not only on the total charge formed by the initial ionizing event but also on the size of the avalanche. In the case of proportional counters, the size of the avalanche can usually be quite accurately controlled. The control is achieved by two factors, (1) a known and **controlled voltage**, and (2) by using **cylindrical geometry**, i.e., by using the cylinder and wire combination mentioned above.

The importance of voltage control is obvious. However, the effects of cylindrical geometry are also of importance. The point is that around the central wire the field varies inversely as the radius, being very high right at the surface of the wire, and falling off rapidly to much lower values farther out in the cylinder. On the other hand, should one use two parallel plates as electrodes, the field between them would be uniform, and the size of the avalanche would depend on where between the plates the avalanche started. With the cylindrical geometry, the avalanche will always start at the same place, the radial distance out from the wire at which the field becomes sufficient to initiate the cascade. Thus in the case of cylindrical geometry an electron will, for a given potential, always produce an avalanche of constant size.

Quantitative considerations may now be introduced. The amplitude of the pulse on the central wire V in volts is related to the charge Q formed in the initial ionizing event through the equation

$$V = \frac{AQ}{C} = \frac{ANe}{C} \quad (10-1)$$

where the charge Q is measured in coulombs and the distributed capacity of the collecting electrode and components attached to it electrically, designated by C , is measured in farads. The quantity A is the so-called **gas amplification**, or size of the avalanche, and is defined as the original electron plus the number of additional electrons produced by each original electron. The original charge produced Q is evidently equal to the number N of electrons formed in the initial ionizing event multiplied by the charge on each e in coulombs. N and A are pure numbers.

Example. Consider a **proportional counter** exposed to a flux of particles containing some **alpha particles** and some **beta particles**. Suppose that the alpha particles are able to penetrate the window, enter the sensitive volume, and there have a residual range resulting in the liberation of 10^5 ions. Since it requires between 30 and 35 ev of energy to form each ion pair, this means that, after the alpha particle has reached the sensitive volume, we are assuming that it has about 3 to 3.5 Mev of energy left. For e we shall take the figure 1.6×10^{-19} coulomb. Assume that the distributed capacity of the central wire is $10 \mu\text{f}$ (10^{-11} farad), and suppose the counter is operated at an A of 1,000. Then the pulse V will be 1.6 volts in amplitude. If we further suppose that the beta particles entering the counter will produce 100 ion pairs each, then the beta pulses will have an amplitude of 1.6×10^{-3} volt. It is quite evident that an electronic circuit can readily be designed which will differentiate between the two differing pulses and will count either or both or, with the aid of discriminating channels, each simultaneously.

The quantity A is related to the quantity called the **first Townsend coefficient** in gas-discharge terminology. It is quite critically dependent upon the voltage across the counter and is a function of the wire radius, the cylinder radius, the pressure, and the kind of gas in the counter. For a derivation of the formulas pertaining to this quantity, see the current texts (Loeb, L. B., "Basic Processes of Gaseous Electronics," University of California Press, 1956) on counters and on gas-discharge phenomena.

To illustrate the dependence of A on voltage, curves are presented in Fig. 10-27 showing measured values of A . Note that in a certain voltage interval the curve may, to good approximation, be described as a straight line on the semilogarithmic plot. Also note that at low voltages, where the value of A approaches unity, the approach is asymptotic. At higher values of A , above 1,000, the dependence on voltage may become quite steep. In terms of the gas-discharge mechanism, this steepening is caused by the appearance of photons in the discharge, which through the photoelectric effect cause still more electrons to be added to the avalanche.

Further, the slopes of the curves of A vs. voltage are steeper for some gases than for others. The curve for **pure argon** is much steeper than, for example, for **meth-**

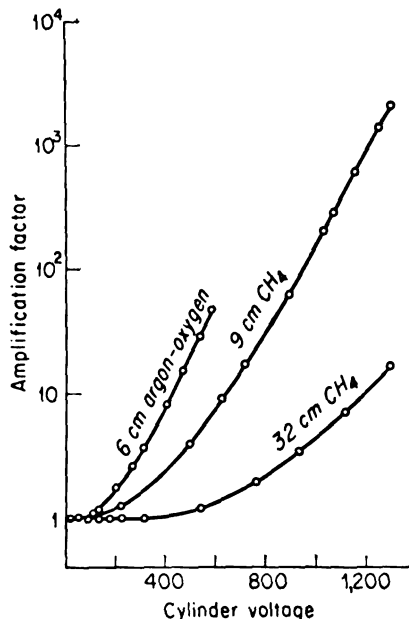


FIG. 10-27. Curves of A as functions of V . Amplification factor (logarithmic scale) against cylinder voltage for 6-cm argon-oxygen ($0.94\text{A} + 0.06\text{O}_2$) and two pressures of CH_4 . (Rose and Ramsey, *Phys. Rev.*, vol. 61, p. 199, 1942; from Koff, S. A., "Electron and Nuclear Counters," 2d ed., Van Nostrand, Princeton, N. J., 1955.)

ane. In general the simpler gases, the monatomic gases and many of the diatomic gases and mixtures thereof have steeper curves than do the polyatomic gases. This effect again is according to Korff to be ascribed to the formation of photons which in turn produce still more electrons. The monatomic gases are in general more transparent to such photons, and also more photons are formed in collision processes. With the polyatomic gases, since they have a much more complex structure of energy levels, radiative unloading of excited states formed in the discharge is less probable, and the supply of photons is smaller. In addition, such photons as

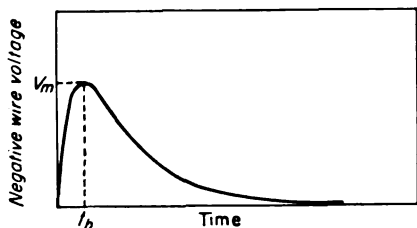


FIG. 10-28. Shape of a pulse. A typical voltage pulse from a self-quenching counter. As viewed on an oscilloscope with a linear sweep, the voltage across R during a pulse is shown here. (After Stever, *H. G.*, *Phys. Rev.*, vol. 61, p. 39, 1942; from Korff, *S. A.*, "Electron and Nuclear Counters," 2d ed., Van Nostrand, Princeton, N. J., 1955.)

are formed have a greater chance of being absorbed by polyatomic molecules, and therefore less chance of reaching the cylinder and there producing photoelectrons. The practical result of this situation is that **polyatomic gases** are often used in proportional counters since this use makes the dependence of A on voltage less critical.

If the wire of such a counter is connected to an oscilloscope, the **pulses** obtained from a proportional counter may be studied in detail. At slow sweep rates, we shall see pulses, the height of which is proportional to N , the number of ions formed in the initial ionizing event. If now the sweep of the oscilloscope is speeded up, so that it re-

quires only a few microseconds to cross the screen, then it will be seen that the pulses have the shape shown in Fig. 10-28.

It will be seen that a pulse is characterized by a fast rise to a maximum value followed by a much slower recovery. Ordinarily the **rise time** of the pulse is of the order of a microsecond or a few microseconds, and the **recovery time** is much longer, in some cases as much as 100 or more microseconds. However, differing from the action of Geiger counters, which are discussed elsewhere in this section, if a second ionizing event should occur very shortly after the first, and before the wire potential had completely recovered, there would be no **dead time** as observed with Geiger counters, and which lasts for 100 or 200 μsec . The so-called **dead times** and **resolving times** of proportional counters are much shorter, by perhaps a factor of 100, than with Geiger counters. The exact amounts and the factors upon which these quantities depend are not today accurately known, but the problem is under investigation.

In terms of the atomic mechanism of the discharge itself, the rise time of the pulse is determined by the time of the formation of the avalanche, its collection upon the central wire, and then both the later stages of the rise; and the entire recovery is determined by the time required for the heavy positive ions formed in the avalanche to drift out to the cylinder and there to be neutralized. We may further point out that, if the circuit connecting the central wire to the oscilloscope is loaded up with resistance or capacity or both, then the RC time to this circuit may become the

determining factor. However, the *RC* time of the circuit can be controlled and should be kept at a low value. Electron-travel times in counters have been measured, and it is known that the **drift velocity** is of the order of 10^7 cm/sec. The time required to build up the avalanche is also short, since the action takes place in the high-field region adjacent to the wire. Because of the short mean free paths, the entire avalanche mechanism will take place in less than the last millimeter next to the wire. The **residual positive ions**, on the other hand, will drift out and must cross the entire counter. At the usual drift velocities this may require of the order of 100 to 200 μ sec, and this effect is therefore the factor controlling the time required for the potential of the wire to return to the value it had before the event took place.

As with any subject which has grown and evolved and which has seen many different workers, often in different countries, contribute to its development, **terminology** was at one time extremely confused. To bring some order into this situation, a series of definitions was agreed upon by the Radiation Counter Tube Subcommittee of the American Institute of Electrical Engineers and the Institute of Radio Engineers. The complete list may be obtained in reprinted form, designated 52 IRE 7.S2 (Testing Procedures) and 52 IRE 7.S3 (Definitions) at \$0.75 and \$0.50, respectively, from the IRE, 1 East 79th Street, New York 21, N.Y.

Historically, proportional counters were at one time called "Geiger-Klemperer counters," to distinguish them from the nonproportional, or "Geiger-Müller counters." Today the term "Geiger-Klemperer" seldom appears in the literature. In the list of standardized definitions, the term **radiation-counter tube** is used. The three words are deemed necessary (1) to identify the nature of what is being counted, (2) to distinguish them from mechanical counters, and (3) to suggest that the devices are in the general category of gas-discharge devices. However, it is often permissible, as we have done in this section, to use the shorter term "counter" when these devices alone are described. A standardized symbol, conforming to the vacuum-tube conventions, has also been proposed and is shown in Fig. 10-29.

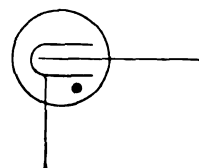


FIG. 10-29. Symbol for radiation-counter tube. (From Korff, S. A., "Electron and Nuclear Counters," 2d ed., Van Nostrand, Princeton, N. J., 1955.)

Terminology and Definitions

The terms and definitions applicable in the case of proportional counters follow.

Avalanche: A cascade multiplication of ions.

Count (in a radiation counter): A single response of the counting system. *See also Tube count.*

Counting rate vs. voltage characteristic: Counting rate as a function of applied voltage for a given constant average intensity of radiation.

Initial ionizing event: An ionizing event that initiates a tube count.

Ionizing event: Any interaction by which one or more ions are produced.

Proportional counter tube: A radiation-counter tube designed to operate in the proportional region.

Proportional region: The range of applied voltage in which the gas amplification is greater than unity and is independent of the charge liberated by the initial ionizing event. *Note:* The proportional region depends on the type and energy of the radiation.

Radiation: In nuclear work, the term is extended beyond its usual meaning to include moving nuclear particles, charged or uncharged, and electrons moving with sufficient speed to enter into nuclear processes.

Radiation counter: An instrument used for detecting or measuring radiation by counting action.

Region of limited proportionality: The range of applied voltage below the Geiger-Müller threshold, in which the gas amplification depends upon the charge liberated by the initial ionizing event.

Sensitive volume (of a radiation-counter tube): That portion of the tube responding to specific radiation.

Tube count: A terminated discharge by an ionizing event in a radiation-counter tube.

A proportional counter, when exposed to radiation consisting of a mixture of various entities which produce ionization, will deliver pulses the amplitudes of which are proportional to the amount of ionization produced within the counter by the entities being counted. In order to make measurements with proportional counters that are meaningful, special circuiting is necessary. The absolute size of the pulse is a function not only of the amount of ionization being produced, but also of the size and shape of the counter, i.e., the radii of the wire and cylinder, the nature and the amount of the gas in the counter, and the voltage applied to the counter. Usually the voltage is the quantity most easily variable; so that by adjusting the voltage the size of the pulse is adjusted to the desired value.

In many applications, **mixed radiation** is present, and it is often desired to measure only one or two components thereof. For example, it may be desired to measure **alpha particles** and also **beta particles** in the presence of a substantial background of **gamma rays**. To do this, **two channels** in the attached electronic circuits are necessary. One channel should be adjusted to measure only the largest pulses from the alpha particles, the second set to accept the beta pulses. The second will also count alpha particles unless these are rejected by additional electronic circuits. The counts due to the beta particles will have a considerable spread in size, and it will be desirable to prove by test that the particles being detected are those desired. This can generally be accomplished by placing thin filters between the source and the detector. A quite thin filter will cut out the alpha particles entirely; and a slightly thicker one will cut out the betas but will not affect the gamma rays much. A block diagram of the required components is shown in Fig. 10-30.

If the counter is to be used not only for **counting**, but also for the **identification of particles** present in the radiation, then it must be calibrated. In the **calibration of a counter** the procedure allows radiation of known characteristics to impinge upon the window. For example, if it is desired to identify an entity which produces a relatively large amount of radiation, the calibrating particles might well be alpha particles. If one starts with fairly long-range alpha particles, such as the 8.3-cm alphas of ThC' , then by the interposition of filters or by moving the source away in air it is possible to diminish the residual range in the counter itself, and hence the

size of the pulse. In this way it may be possible to bracket the size of the pulse produced by the unknown particle, and thus to determine the amount of energy which this unknown dissipated as ionization. If the charge or the momentum of the unknown is also known by other measurement, then its mass can be identified. To summarize: a proportional counter counts the number of particles which have

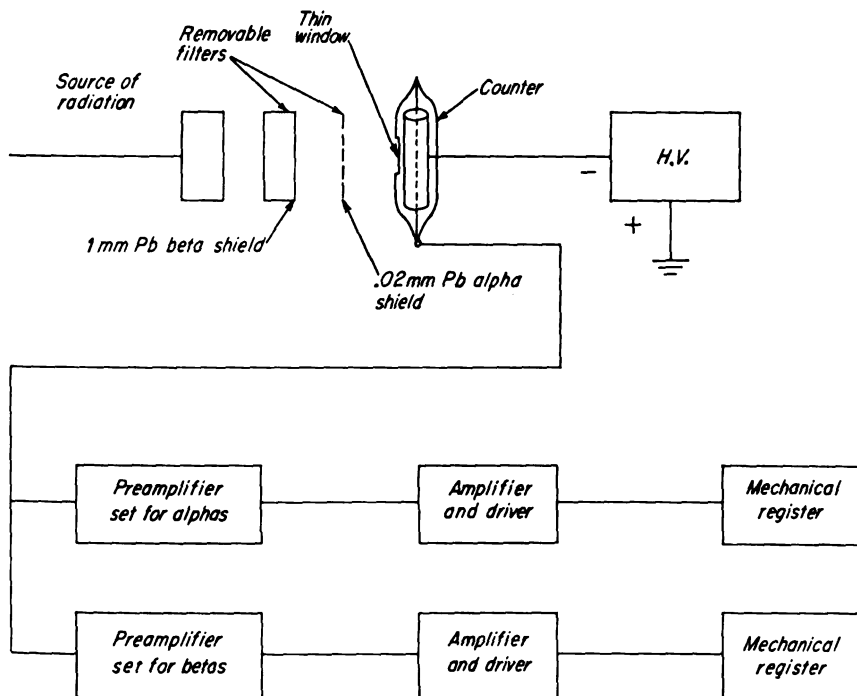


FIG. 10-30. Block diagram of counter and discriminators for mixed radiation. Arrangement for simultaneously counting alpha particles in presence of large beta and/or gamma background.

enough energy to produce one or more ion pairs. Further, a proportional counter can allow a determination to be made of the amount of ionization produced along that part of the range which lies within its sensitive volume. Thus, for **mixed radiation**, or for measurement of, for example, a small number of alpha particles in the **presence of a large background** of other radiation, proportional counters are especially well suited. They should be followed by electronic circuits which enable pulse selection to be made, since by **discriminating circuits** it is possible to select pulses of certain size ranges only and to reject if necessary a much larger number of other pulses due to other entities which for this experiment may be considered "background." Proportional counters will measure the number of particles and the ionization produced by each.

Typical Problems

A proportional counter, operating at a gas amplification of 100, detects alpha particles. The distributed capacity of its wire collecting is $15 \mu\text{f}$. Take n as 10^6 . Find the size of the pulse to be expected on its wire. *Ans.:* From Eq. (10-1), the size of the pulse will be 0.106 volt.

A proportional counter is observed to give pulses of 10^{-2} volt amplitude for certain particles. If the counter is operated at an A of 1,000, and if C is $15 \mu\text{f}$, find the amount of ionization liberated by each particle. *Ans.:* From Eq. (10-1), 937 ion pairs.

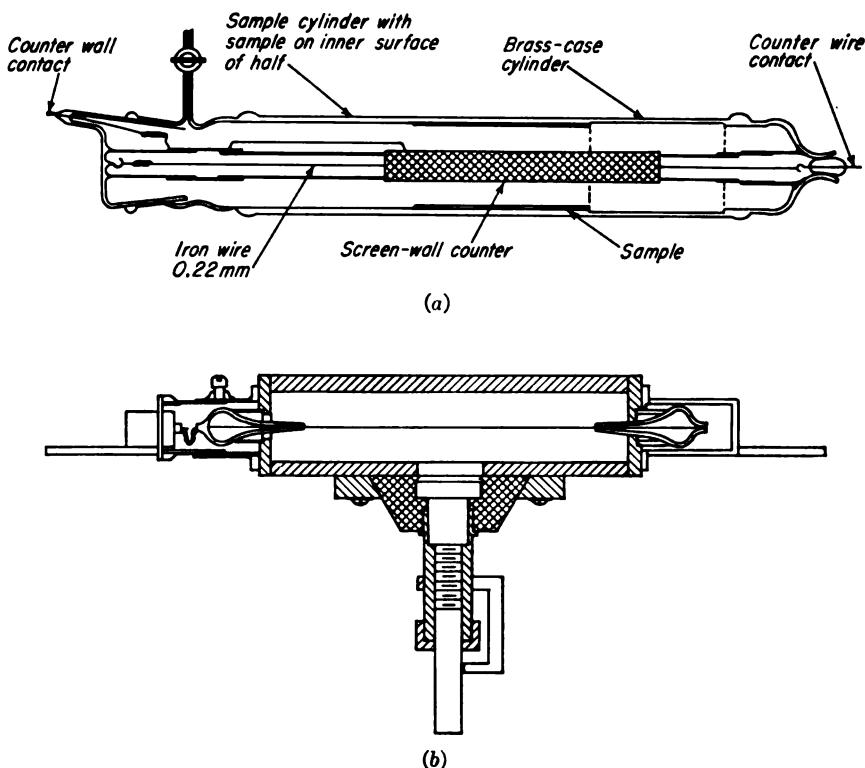


FIG. 10-31. Counters for special purposes. (a) Screen-wall counter assembly. (From Libby, W. F., et al., *Rev. Sci. Instr.*, vol. 22, p. 225, 1951.) (b) Cross-section view through plane AA' of assembled gas-flow proportional counter. Spindle and sample platform are shown in lowered position.

An example of a proportional counter made for a special purpose is a **neutron counter**. If a counter is filled with a gas containing boron, such as the gas BF_3 , and is exposed to a flux of slow neutrons, then neutrons passing through the counter may be captured by the boron nuclei, which for this isotope B-10 have a strong capture probability for slow neutrons. The nucleus then splits into two unequal fragments,

He and Li. These two fragments separate with considerable energy and can liberate an appreciable amount of ionization in the gas.

Such an arrangement immediately lends itself to proportional-counting techniques. By proper discriminator circuits, rejecting small pulses, a neutron counter can detect a small number of neutrons in the presence of a large background flux of gamma radiation. The details of neutron detection are treated in other chapters, and this is cited here merely as an example of a proportional counter adapted for a special problem.

Another example of a **proportional counter for a special purpose** is the counter used by Libby (Libby, W. F., *et al.*, *Rev. Sci. Instr.*, vol. 22, p. 225, 1951) for the determination of radiocarbon (C-14) in archaeological specimens. A drawing of this counter is shown in Fig. 10-31a. In this case, the carbon to be analyzed is converted into a solid and painted onto the inside of the cathode, so that the electrons originating from the disintegration of the C-14 will be able to enter the sensitive volume. The counter can in this case be filled with any convenient gas. The one-level counting possible with this design comes about because of the good shielding arrangements. In the actual use, the counter itself is shielded by several centimeters of lead, which has the effect of greatly reducing the counting rate because of gamma rays originating from local contamination. Further, a set of anticoincidence counters arranged around the outside eliminates the majority of the counts due to cosmic rays.

Another example of proportional counters for special purposes is the counter developed by Simpson (Simpson, J. A., *Rev. Sci. Instr.*, vol. 18, p. 1884, 1947; vol. 19, p. 733, 1948) in which methane at atmospheric pressure is allowed to flow through the counter. The counter is shown in Fig. 10-31b. Still another example is the methane proportional counter developed by Hurst (Hurst, G., *et al.*, *Nucleonics*, vol. 12, no. 8, p. 36, August, 1954). In this design he uses an ethylene-filled proportional counter to detect fast neutrons, making use of the protons recoiling as a result of fast neutron impact.

GEIGER COUNTERS

This discussion is devoted principally to Geiger-Müller counters, although many of the basic principles are the same as those for proportional counters. The Geiger counter is a device of similar nature to a proportional counter but working in a different voltage interval. We assume a cylinder, with a wire along its axis as in the case of the proportional counter, and that the whole is placed in a suitable envelope, so that it may be filled with a known gas at a desired pressure. The same device which operated as a proportional counter may very often operate as a Geiger counter, if only the voltage is readjusted.

Consider therefore a counter, and let the voltage be such that it is operating in the proportional region. Then if the voltage is increased, the output pulses will get larger and larger. However, a change in the characteristics of the pulse also occurs, in turn because of the operation of a part of the discharge mechanism which is quite unimportant in the proportional region. The result of this operation, which we shall discuss in detail below, is that the discharge spreads along the central wire. When the entire length of the wire is involved, the size of the pulse no longer

is a function of the number of ions formed in the initial ionizing event. Thus, a single electron, or an ionizing event in which a hundred thousand electrons are formed, will each initiate a discharge. The size of each avalanche is not larger in one case than in the other. The device is under such circumstances said to be operating in the Geiger region, and is called a Geiger counter. The pulses which such a counter produces are usually substantially larger than those produced by proportional counters. Moreover, the device has lost its property of being able to distinguish between the entities initiating the discharge, and any ionizing event merely functions as a trigger. The Geiger counter therefore counts ionizing events, regardless of their size.

After an initial ionizing event has taken place, the first part of the subsequent **gas-discharge mechanism** is the same as described above for proportional counters. The electrons formed in the event drift in toward the central wire and, entering the high-field region, produce still additional electrons by collision. An avalanche is built up. Since the high-field region is only a small fraction of a millimeter about the central wire, the first avalanche itself is of finite size. However, if the collisions are sufficiently energetic so that photons are formed in the avalanche in appreciable numbers, these photons may initiate additional avalanches through the photoelectric effect. These new avalanches are formed at places along the wire other than those occupied by the previous avalanches. Eventually this process spreads down the length of the wire; so that the entire wire is covered with a thin discharge sheath.

Inside this discharge sheath, the effect of the positive-ion space charge has been shown by the Montgomerys (Montgomery, C. G., and D. D. Montgomery, *Phys. Rev.*, vol. 57, p. 1030, 1940) to cause a lowering of the field sufficient to terminate the avalanche growth. Therefore, once the entire wire is covered, the pulse does not grow larger. The initial event causing this effect may be one electron or any larger number. The size of the pulse is determined by the total avalanche and not by the number of ions originally formed. In terms of Eq. (10-1), the quantities A and N are no longer independent variables. Their product is a constant for any one voltage. If the voltage is increased, the total size of the pulse increases. In other words, AN depends on the voltage. Since e and C are constants of nature and of the instrument, respectively, all pulses will be the same size at any one voltage, regardless of the number of ions initially formed.

As one raises the voltage, starting in the proportional region, the transition into complete Geiger operation is a gradual one. The situation may be understood by referring to Fig. 10-32, in which the various voltage regions are shown. At low voltages A is unity, and one simply collects those ions which were first formed. As the voltage is raised, the avalanche mechanism begins to operate, and A becomes larger than unity but is independent of N . As the voltage is still further raised, the independence of A and N slowly begins to disappear, until it vanishes in the Geiger region.

This is illustrated in Fig. 10-31 by the use of two curves. It is assumed in Fig. 10-31 that two different ionizing events occur. Curve A is the observed voltage pulse upon the central collecting electrode when an alpha particle liberates a large number of electrons in passing through the device. Curve B is for a less ionizing particle, for example, a fast beta ray or a cosmic-ray meson, which in passing through the device leaves only a few ions behind it. Both curves are horizontal at

low collecting voltages, and both bend upward when the avalanche mechanism starts. The voltage at which this multiplication starts is called the **threshold for proportional-counter action**. The voltage region below this threshold is called the **ionization-chambers region**, for in this interval the device is acting as an ion chamber. Above the threshold for proportional counting the next interval is the **region of the proportional-counter action**. Then, as the voltage across the device

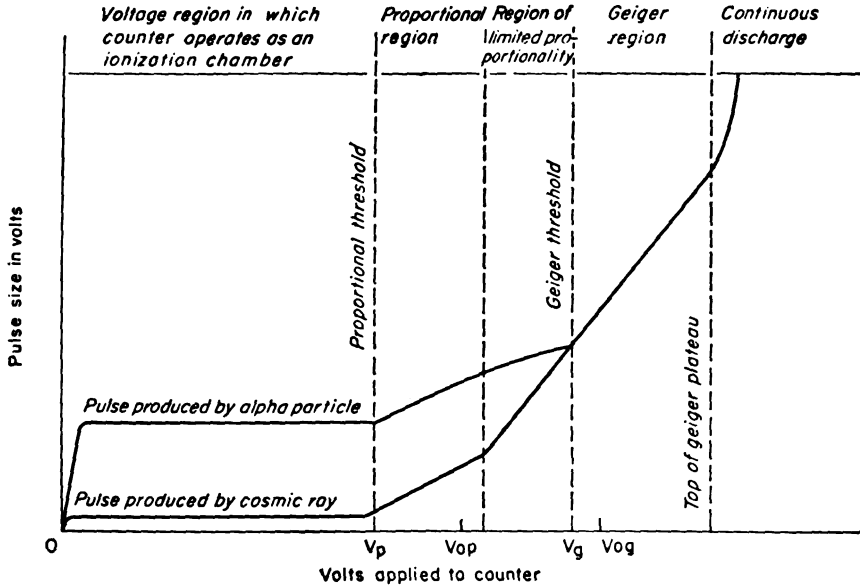


FIG. 10-32. Operating characteristics of a counter in various regions for large and small ionizing events. Top curve shows pulse resulting from passage of alpha particle producing 10,000 ions during its passage through the counter. Bottom curve, pulse resulting from cosmic ray producing 30 ions. The curves merge at the Geiger threshold. Recommended operating voltages: for proportional counting, V_{op} , near top of proportional regions, to get large pulses; V_{og} , for Geiger counting near Geiger threshold to avoid excessive pulse size and damage to counter. (From Korff, S. A., "Electron and Nuclear Counters," 2d ed., Van Nostrand, Princeton, N. J., 1955.)

is further raised, the two curves *A* and *B* start to draw together. The values of *A* characteristic of large pulses are not quite so large as those characteristic of small pulses. In other words, there is a dependence of *A* on *N*. Finally the two curves merge, and the voltage at which this happens is called the Geiger threshold, or the "threshold for Geiger-Müller counter action." The present discussion will deal with devices in the Geiger region.

After the initial avalanche has been formed, the secondary avalanches have come into being, and the discharge has spread along the wire, the positive ions start to move out to the cylinder. The electrons move much faster through a high-field region and have all been collected on the central wire before the positive-ion space-charge sheath has moved appreciably. When the slow-moving positives reach the cylinder they are neutralized. If no further electrons are formed the process then

terminates and the counter recovers and is ready for the next count. However, if one or more electrons are formed during the neutralization process these new electrons will again drift toward the wire under the field and as soon as they enter the high-field region will form new avalanches. In this event the initial discharge is not terminated, and the counter goes into what is called a **continuous discharge**.

Terminology and Definitions

As earlier, we list here the terms officially defined in the IRE standards and applicable to Geiger counters.

Background counts: Counts caused by radiation coming from sources other than that measured.

Count (in a radiation counter): A single response of the counting system. *See also Tube count, page 10-38.*

Dead time: The time from the start of a counted pulse until an observable succeeding pulse can occur. *See also Recovery time.*

Efficiency (of a radiation-counter tube): The probability that a tube count will take place with a specified particle or quantum incident in a specified manner.

Externally quenched counter tube: A radiation-counter tube that requires the use of an external quenching circuit to inhibit reignition.

Gas-filled radiation-counter tube: A gas tube, in a radiation counter, used for the detection of radiation by means of gas ionization.

Geiger-Müller counter tube: A radiation-counter tube designed to operate in the Geiger-Müller region.

Geiger-Müller region (of a radiation-counter tube): The range of applied voltage in which the charge collected per isolated count is independent of the charge liberated by the initial ionizing event.

Geiger-Müller threshold: The lowest applied voltage at which the charge collected per isolated tube count is substantially independent of the nature of the initial ionizing event.

Normalized plateau slope: The slope of the substantially straight portion of the counting rate vs. voltage characteristic divided by the quotient of the counting rate by the voltage at the Geiger-Müller threshold.

Overvoltage: The amount by which the applied voltage exceeds the Geiger-Müller threshold.

Plateau: The portion of the counting rate vs. voltage characteristics in which the counting rate is substantially independent of the applied voltage.

Plateau length: The range of applied voltage over which the plateau of a radiation-counter tube extends.

Predissociation: A process by which a molecule that has absorbed energy dissociates before it has had an opportunity to lose energy by radiation.

Quenching (in a gas-filled radiation-counter tube): The process of terminating a discharge in a radiation-counter tube by inhibiting reignition.

Recovery time (of a radiation counter): The minimum time from the start of a counted pulse to the instant a succeeding pulse can attain a specific percentage of the maximum value of the counter pulse.

Reignition (of a radiation-counter tube): A process by which multiple counts are

generated within a counter tube by atoms or molecules excited or ionized in the discharge accompanying a tube count.

Relative plateau slope: The average percentage change in the counting rate near the mid-point of the plateau per increment of applied voltage. *Note:* Relative plateau slope is usually expressed as the percentage change in counting rate per 100-volt change in applied voltage.

Resolving time (of a radiation counter): The time from the start of a counted pulse to the instant a succeeding pulse can assume the minimum strength to be detected by the counting circuit. (This quantity pertains to the combination of tube and recording circuit.)

Self-quenched counter tube: A radiation-counter tube in which reignition of the discharge is inhibited by internal processes.

Spurious tube counts (in radiation-counter tubes): Counts in radiation-counter tubes other than background counts and those caused by the source measured.

The process of terminating the discharge is called **quenching**. There are two fundamentally different ways in which this result may be achieved. The first is produced by external causes. It is possible to connect the counter to electronic circuits which have the effect of reducing the voltage across the counter, after the count occurs, for a time long enough to permit the electrons formed in the neutralization process to be collected without producing avalanches. To do this, it is not necessary to reduce the voltage across the counter to zero, but merely to a value in the lower part of the proportional region. This can be done by a variety of devices, which are called **quenching circuits**. Counters requiring the use of such circuits are called **externally quenched counters**.

The alternatives to such counters are those in which, because of an internal mechanism, no secondary electrons are formed by the various processes which take place when the positive ions are neutralized. Counters which exhibit this property are called **internally quenched counters**, or **self-quenched counters**. In general it may be said that counters filled with monatomic gases, or with the commoner diatomic gases or mixtures thereof, are usually not self-quenching; whereas those filled with polyatomic gases usually do exhibit self-quenching properties. The first account of quenching gases in the literature was given in about 1937 by Trost (*Z. Physik*, vol. 105, p. 399, 1937).

The details of the **self-quenching mechanism** were studied by Korff and Present (*Phys. Rev.*, vol. 65, p. 274, 1944), who presented the following picture of the events. Two processes are involved. Consider a counter filled with argon plus a quenching vapor such as alcohol or ethyl acetate. After the avalanche is over there will be positive ions of argon and of the organic vapor which move toward the cylinder. Because of collisions with neutral vapor molecules, opportunity will be present for electron transfer to take place. If the ionization potential of the organic vapor is below that of the argon, which is almost always the case, an argon ion colliding with a vapor molecule will leave the neutral argon atom and an ionized molecule. Thus by the time the positive ion sheath reaches the cylinder it will be almost 100 per cent organic vapor ions. Now when such complex ions are neutralized they ordinarily do not radiate, but lose their excess energy by predissociation. There being no photons reaching the cylinder, no secondary electrons will be formed. The supply of electrons being thus shut off, the discharge is terminated.

Another factor operative in the atomic mechanism inside counters is the production of photons in the avalanche itself. If photons produced in the avalanche produce photoelectrons at the cylinder, then an additional supply of electrons is available from this source. However, in the counters in which organic vapors are present, strong absorption usually takes place of those photons which are energetic enough to produce photoelectrons, i.e., which have energies greater than the photoelectric work function of the material of which the cylinder is made. Many organic vapors have strong absorption bands in the far ultraviolet. The ultraviolet absorp-

tion spectrum plays an important role in the quenching mechanism.

The pulse produced by a Geiger counter is shown for a typical case in Fig. 10-28. It will be seen that it consists of a rapid rise followed by a much slower recovery. The time required for the pulse to build up is determined by several factors, the longest of which is usually the time taken for the discharge to spread down the length of the counter wire. The velocity with which this discharge spreads has been measured by a number of observers and has been found to be of the order of 10^7 cm/sec, the exact values depending on the voltage, gas type, and pressure, and possibly on wire radius. Thus for a counter 20 cm long we might expect that it will require about $2 \mu\text{sec}$ for the discharge to form a complete sheath along the entire length of the counter. The recovery is much

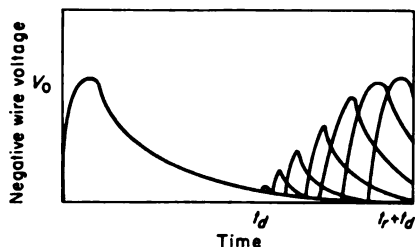


FIG. 10-33. Stever dead-time diagram. Drawing of oscilloscope pattern of the result of the dead-time experiment. This shows the dead time t_d and the recovery time t_r . The time $(t_d + t_r)$ corresponds to the arrival of the positive-ion sheath at the cylinder. The time t_d corresponds to the point in the transit of the positive ions at which the field about the wire has returned to threshold field. (From Stever, H. G., *Phys. Rev.*, vol. 61, p. 40, 1942; from Korff, S. A., "Electron and Nuclear Counters," 2d ed., Van Nostrand, Princeton, N. J., 1955.)

slower, for this is controlled by the motion of the positive ions. Simple calculations agree with the measured values, namely, about 100 to 200 μsec for the recovery process.

If a counter received a second ionizing event during the first part of the process, no new pulse will be formed. Geiger counters exhibit definite "dead times," which may be illustrated in Fig. 10-33. In this figure we show the pulses found on the wire when second pulses follow at various times after a first one. Since it is possible to trigger an oscilloscope sweep with a pulse, we can use this synchroscope action to exhibit **dead times** and **resolving times**. Let a counter be connected to such a synchroscope and be exposed to a strong flux of radiation. By a strong flux we mean that the average time between ionizing events is much smaller than the "dead time" of the counter. In this case, each time the counter discharges, another discharge will follow very shortly, and this second discharge usually will take place before the counter is fully recovered. The envelope of the entire pattern on the oscilloscope screen allows two quantities to be defined. The "dead time" will be seen to be that time after the first event during which the counter is insensitive to further events. A pulse arriving just after the dead time will not be of full size. After a somewhat

longer interval, the **recovery time**, the counter recovers so that the subsequent pulses are of full size. The original picture of this action (shown in Fig. 10-32) was given by Stever (Stever, H. G., *Phys. Rev.*, vol. 61, p. 38, 1942) and is sometimes called a **Stever diagram**.

It will be seen from the discussion above that a Geiger counter is a device which counts the number of ionizing events which take place inside it. Such a device is not capable of distinguishing between ionizing particles in the sense that the pulses are dependent only upon the number of ionizing events and not upon the amount of ionization formed in each event. Geiger counters will measure the relative radiation outputs of sources, provided that the nature of the radiation is kept constant. They can also make absolute counts, provided careful consideration is given to the geometrical arrangements, and provided the efficiency of the counter for the particular kind of radiation is known.

The **efficiency** of a counter in detecting radiation is defined as the probability that a count shall take place when the radiation to be detected enters the sensitive volume. In terms of the atomic mechanism, since a single initial electron can start an avalanche, it will suffice that a single electron be produced in the initial ionizing event. For the case of charged particles, the probability can be calculated. The formula for the efficiency of a counter counting charged particles is

$$G = 1 - e^{-slp} \quad (10-2)$$

where the efficiency G is a number smaller than unity, expressible in per cent by moving the decimal point over two places, e is the base of natural logarithms, s is the specific ionization of the particle to be detected, in ions per centimeter at standard temperature and pressure (STP), p is the pressure of the gas in the counter, and l is the average path of the entity to be detected in passing through the counter.

Example. Consider a Geiger counter counting cosmic-ray mesons. Suppose that this counter is filled to a pressure of 15 cm Hg (about 0.2 atm) of argon. For argon the values of s are known for such particles, about 30 ions/cm at STP. Suppose the path followed by the average meson through the counter is 2 cm. Then the quantity slp is 6, and since e^{-6} is 0.0024, G is 99.76 per cent. For heavy particles such as alpha particles or slow protons, s is so large that G is indistinguishable from unity, and the counter may be said to be always 100 per cent efficient. For nonionizing entities such as neutrons or gamma rays the situation is much more complex. The efficiency of counters for neutrons will be discussed. The efficiency for gamma rays we shall discuss in the paragraph below where gamma-ray counters are described.

In the case of a self-quenching counter, the principal factor affecting the **useful life** is the decomposition of the quenching gas. The act of quenching causes some molecules to decompose each time a count occurs. The following list of recommendations is therefore given with a view of securing longest useful life.

Do not operate the counter above its rated voltage. If the counter should at any time go into a "continuous discharge" disconnect the voltage immediately. Never leave the high voltage connected to the counter when the device is not in use.

The manufacturer often specifies the number of counts for which he guarantees the counter. These data should be kept in mind while using the device.

In designing counters, those gases which contain molecules which can decompose more than once and still quench may give longer lives.

In the case of the externally quenched counters, the factors determining the life are quite different. However, here again it is important not to allow excessive currents to flow. Large currents may cause pitting or otherwise damage the collecting electrode. The appearance of a single microscopic sharp point on this electrode may be sufficient to cause the counter to show spurious discharges.

Certain gases polymerize in the discharge. The resultant heavy material is often deposited under the action of the field on the central wire. In certain cases this effect can determine the life of the counter, for a counter will not work properly when the wire has an appreciable amount of nonconducting matter on it.

The halogen-quenched tubes can have their lives shortened by excessive currents. In general they are not subject to the limitation produced by decomposition.

Counters may go bad in the normal process of use, but in some cases can be rejuvenated. The procedure depends on correct diagnosis as to the cause of the trouble.

If the counter "goes bad" owing to progressive decomposition of the quenching constituent, the situation can be rectified by evacuating the gas and refilling with fresh gas plus quenching agent. If the trouble arises from the deposit on the central wire of a polymeric product formed in the discharge, the remedy in some cases is to pass a current through the wire, heating it white hot to drive off any surface layers. This procedure depends on having both ends of the wire electrically available. Replacing the wire is another alternative in this case.

Replacing the wire (and then refilling) is the remedy if the counter shows an excessive number of spurious counts because the wire has become pitted by too high currents.

Small amounts of the halogen gases will work as quenching agents. **Halogen-quenched counters** normally contain a major part of some inert gas such as **argon**, to which a small amount of **neon**, and a small amount of one of the halogens, **chlorine** or **bromine**, has been added. Various differing mixtures have all been found to work. Other inert gases, such as **helium** or **krypton**, will also work, but krypton is usually too expensive and helium has the disadvantage of having a low specific ionization and therefore requiring more gas for a given efficiency. Other halogens also work, but fluorine is so active chemically as to present great manufacturing difficulties and iodine has so low a vapor pressure at normal temperatures as to cause trouble on this score. Since the halogens are all active chemically, halogen-quenched counters are made with inert materials such as stainless steel for electrodes.

Halogen-quenched tubes have the advantage of longer life, since the halogen decomposes but recombines back into its original diatomic state. Hence the life of the tube is not controlled by the progressive decomposition of the quenching constituent. However, these tubes have two disadvantages, one being that the plateaus are not so flat as obtainable with conventional fillings, and the second that negative-ion effects may be present, and longer time lags may be found.

It has been shown by Maze (Maze, R., *J. phys. radium*, vol. 7, p. 164, 1946) that it is possible to make counters which use an **external cathode** (outside the glass).

In this construction, a special soft glass is used. The cathode takes the form of a painted coating of some conducting material such as aquadag on the outside. Such counters have excellent self-quenching characteristics, long lives, and long flat plateaus. The main disadvantage is in the fact that the glass becomes polarized, and the counter is limited in its speed of operation by the long time needed for the ions to diffuse through the glass.

In all the above discussion, it has been assumed that some sort of **window** is provided so that the radiation to be detected can reach the sensitive volume. For gamma rays or X-radiation or cosmic rays, there is no problem on this score, as the radiation can easily penetrate the walls of a normal counter. For soft beta rays or for alpha radiation, the problem becomes much more difficult mechanically.

For soft radiation it is necessary to provide some sort of window. Windows may take two general forms, with many possible modifications of each. The two forms are (1) the thin-wall type and (2) the special window.

In the first, or **thin-wall window**, the cylinder is made quite thin. Such counters can be made in two ways, (1) by drawing out a glass tube until its walls are thin and then painting a conducting cathode surface on the inside with some conducting substance, e.g., aquadag, or evaporation copper or other metal on it, or chemically silvering it; and (2) by machining the cathode itself and using this as the envelope. Counters made in this way can readily be made with the wall thin enough to admit beta rays, and are often called beta-ray counters. Since these walls are also rather fragile, it is usual to protect them by having a heavy perforated shield outside, to act as mechanical protection while the radiation to be detected passes through the perforations.

In the **special-window** types, the normal wall is used, and an aperture is provided with an extremely thin covering. For example, a normal counter may be equipped with a very thin end, a make of mica or glass or metal or plastic through which the radiation passes. The problem is to provide enough mechanical strength for the window so that it will resist the pressure differences between the atmosphere and the lower inside pressure. Such windows can be made the equivalent of 1 cm of air in thickness, about 1 mg/sq cm in material. Alpha particles can penetrate such windows. Mechanical support for such thin windows must be provided, sometimes in the form of a grid which sustains the pressure and through the interstices of which the radiation can pass. Also care must be taken that nothing from the outside shall actually touch the window for fear of breaking it.

Figure 10-34 shows some typical thin-window types.

The **time lag** of a counter is defined as that time interval between the initial ionizing event and the time at which a detectable voltage change has appeared on the collecting-electrode system. The time lags of various counters under many conditions have received extensive study, and the principal factors operative will be discussed here.

The longest time lags usually found are those produced by negative ions formed in the counter. If the electron (or electrons), formed in the initial ionizing event should be captured to form a negative ion at any time before it has drifted into the high-field region near the central wire, its drift velocity will be diminished by factors of as much as 1,000. Such effects have been observed in counters containing electro-negative gases such as oxygen or water vapor or the halogens. In some such cases

the negative ion may require many microseconds, even as many as $100\ \mu\text{sec}$, to reach the high-field regions and there to initiate an avalanche.

If, on the other hand, negative-ion formation does not take place (and it is easy to arrange things so that it will not, by using gases which have very small electron-capture cross sections, such as argon, hydrogen, and other gases), then the time lag is determined by the sum of three processes. First is the length of time needed for the electron to drift into the high-field region, second is the time needed for the avalanche to build up, and third is the time needed for enough electrons to be released from the electrostatic-image forces produced by the positive ions, so that the

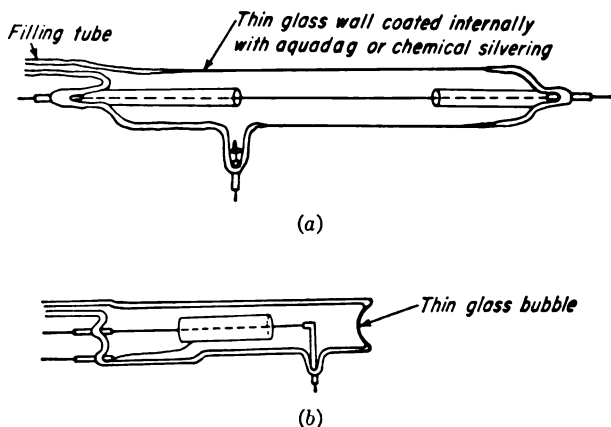


FIG. 10-34. Counters for special purposes. (a) Thin-walled beta-ray counter. (b) Counter with thin end window for alpha particles.

wire potential can start to change by an observable amount. The first of these processes is quite fast, requiring usually of the order of a few tenths of a microsecond. The second part, the avalanche build-up, is still faster, occurring in a few millimicroseconds. The third part of the process is usually the determining factor, since it may partly involve the length of time required for the discharge to spread down the length of the wire. At this point it becomes appropriate to consider the minimum observable voltage change on the wire. This will depend upon the sensitivity of the detecting arrangements: These can be such that the time lags may be of the order of tenths of microseconds or more, or sometimes less.

In the case of **gamma-ray counters**, the efficiency depends upon the probability that the gamma-ray or X-ray photon will produce at least one electron inside the counter. Electrons are produced by photons by several processes. They may be produced in the gas itself by the Compton effect, or by photoionization, or they may be ejected from the walls. For photon energies which are substantially above the normal ionization potentials for the substances used in the gas of the counter, which ionization potentials are 10 to 20 volts for most gases, the cross section for electron production from the gas atoms is small. It is more probable that an electron will be ejected from the walls, either as the photon enters or as it leaves the inside surface of the cylinder.

It has been known for some years that the probability of ejection of electrons from walls of such vessels is a function of the atomic number of the material of which the wall is made. Hence for such counters it has been customary to employ materials such as lead or bismuth for the cylinders.

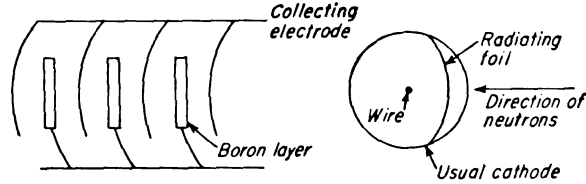


FIG. 10-35. Possible arrangements of internal radiators. (From Korff, S. A., "Electron and Nuclear Counters," 2d ed., Van Nostrand, Princeton, N. J., 1955.)

For a number of applications, including the production of recoil protons for fast-neutron detection, and also the production of additional wall electrons for gamma-ray detection, it is possible to make use of still another procedure to increase the yield of such particles. This increase is achieved by the use of radiating elements, placed inside the counter itself. Such devices are called **internal radiators**, for they form the source of the particles which will be later counted. The net effect of such internal radiators is to increase the area of the surface from which the particles which will initiate the detecting action can emerge. Figure 10-35 shows some typical arrangements.

The possibility of operating Geiger and/or proportional counters in coincidence makes possible a vast extension in detection techniques and assists in securing new kinds of information about complex ionizing events. By **coincidence technique** is

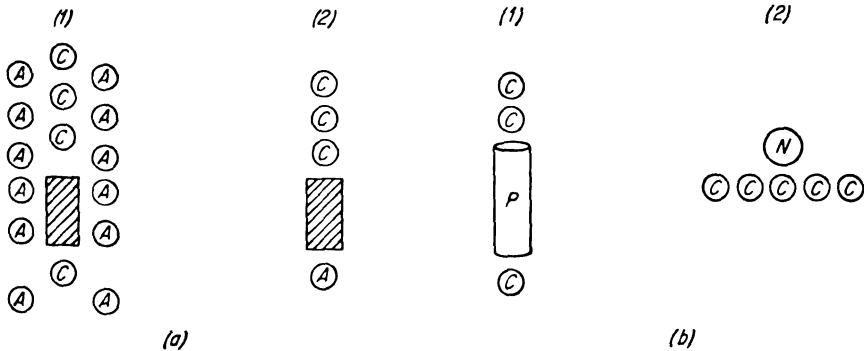


FIG. 10-36. Coincidence arrangements. (a) Typical anticoincidence arrangements. (1) Employs anticoincidence counters A as guard counters to assure that the counters C are not set off by showers from the side. (2) Ensures that a particle having passed through C is absorbed and does not emerge in the direction determined by C. (b) Typical counter arrangements using counters in coincidence with proportional counters. (1) The counters C define a path for particles through the proportional counter P. (2) Coincidences between neutrons N and showers C are counted. (From Korff, S. A., "Electron and Nuclear Counters," 2d ed., Van Nostrand, Princeton, N. J., 1955.)

meant the arrangement of electronic circuits so adjusted that two or more counters must discharge simultaneously in order for an event to be recorded.

In the simplest arrangement, two counters are connected in coincidence, and therefore determine a line, for each coincidence count means that a particle has passed through both. Many more complex modifications are possible and are in general use. Multiple coincidences are readily arranged electronically. Proportional counters and other devices such as Cerenkov counters or scintillation counters (discussed in other sections) can be connected in coincidence with similar circuits. A proportional counter can be placed between two ordinary Geiger counters. If the two Geiger counters discharge, then the particle has followed a line determined by the two **path-defining** or **directional** counters through the proportional one, and this can be used to define a path good enough so that fairly precise ionization measurements may be made. Figure 10-36 shows a number of possible arrangements.

PHOTOGRAPHIC MONITORING OF RADIATION

George M. Corney

References: Hine and Brownell, "Radiation Dosimetry," Academic Press. Ehrlich, "Photographic Dosimetry of X and Gamma Rays," GPO. Mees, "Theory of the Photographic Process," 2d ed., Macmillan.

Photographic films can be used as small, rugged, inexpensive radiation dosimeters which give a permanent record. However, they are not absolute devices, in that the developed density (i.e., the response) for a given exposure depends upon both photographic-processing variables and the quality of the radiation to which they have been exposed. Photographic dosimeters therefore require calibration.

PHOTOGRAPHIC CONSIDERATIONS

The characteristic curve (H and D curve, D-log E curve) of a typical photographic material used in radiation monitoring is shown in Fig. 10-37a. It relates the **density*** of the processed film with the logarithm of the exposure. Exposure may be defined as a quantity of radiation (e.g., as a number of roentgens) or as the product of radiation intensity and time. In some monitoring and film-calibration procedures it is convenient to use relative exposure, i.e., to express exposures in terms of their ratios to some arbitrarily chosen exposure, whose absolute value often need not be known.

For a particular processing technique, the *shape* of the characteristic curve is independent, for practical purposes, of the quality of the exposing X- or gamma

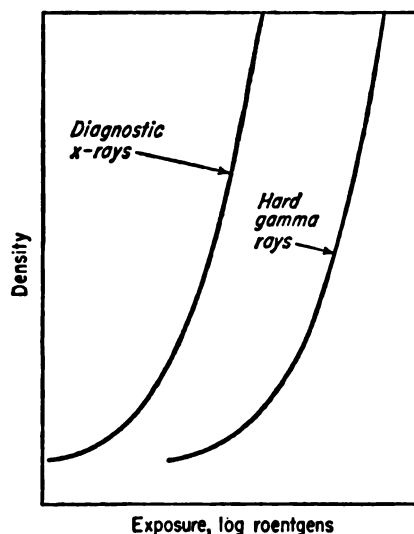


FIG. 10-37a. Characteristic curves of a typical monitoring film when exposed to diagnostic X-rays (left) and hard gamma rays (right). The shapes of the two curves are the same for practical purposes.

* Density = $\log_{10}(I_0/I)$, where I_0 is the intensity of the light incident upon a processed film and I is the light transmitted.

radiation, and for films in which the latent image does not fade, it is also independent of the rate at which the exposure is received. (Wilsey, R. B., *et al.*, Experiments in the Photographic Monitoring of Stray X-rays. I. General Considerations. The Choice of Film-calibrating Radiations in Roentgen Therapy at 220 kvp and 1,000 kvp, *Radiology*, vol. 66, p. 408, March, 1956. Ehrlich, M., and W. L. McLaughlin, Reciprocity Law for X Rays. Part I: Validity for High-intensity Exposures in the Negative Region. *J. Opt. Soc. Am.*, vol. 46, p. 797, October, 1956.) The *location* of the curve along the $\log E$ axis does depend upon radiation quality; hence characteristic curves for a given photographic material made with two different radiation qualities might be displaced from one another along the $\log E$ axis, but the shape of the curves would be the same (see Fig. 10-37a).

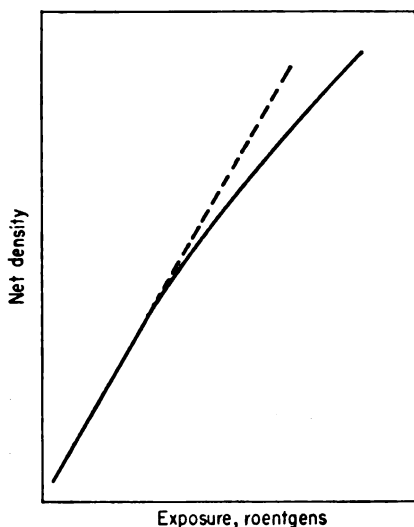


FIG. 10-37b. Net density vs. exposure curve for the typical monitoring film of Fig. 10-37a.

The *shape* of the **characteristic curve produced by beta rays** energetic enough to penetrate the whole film is the same as that for X- and gamma rays and is independent of beta-ray energy. The *location* of the beta-ray curve along the $\log E$ axis is, as is the gamma-ray curve, dependent upon the beta-ray energy (Dudley, R. A., Photographic Detection and Dosimetry of Beta Rays, *Nucleonics*, vol. 12, no. 5, p. 24, May, 1954. Digby, N., *et al.*, Photographic Effect of Medium Energy Electrons, *J. Phot. Sci.*, vol. 1, p. 194, November-December, 1953. Fleemann, J., and F. S. Frantz, Film Dosimetry of Electrons in the Energy Range 0.5 to 1.4 Million Electron Volts, *J. Research NBS*, vol. 48, p. 117, February, 1952. Tochilin, E., *et al.*, Response of Photographic Emulsions to Charged Particles and Neutrons, *Radiation Research*, vol. 4, p. 467, June, 1956).

The **net density* vs. exposure curve** for a typical monitoring film is shown in Fig. 10-37b. Many films, particularly direct-exposure X-ray films, have a linear relation between density and exposure over a limited density range. Such a linear relation, if it is known to exist, may decrease the arithmetical work involved in radiation monitoring. However, before using such a relationship, preliminary tests should be made, since the degree of approximation to a straight line and the range of density over which it exists are a function both of the photographic material and of the development technique. It is particularly important to check the linearity of the net density vs. exposure curve before using any monitoring procedure which involves the difference in the densities produced under two filters placed over the film, e.g., in the monitoring of "mixed" gamma rays or of beta rays in the presence of gamma rays.

* Net density = total density less fog and support density.

The response of photographic films to combinations of X-, gamma, and beta rays may be assumed, for the purposes of practical radiation monitoring, to be independent of the order in which the radiations are received (Solon, L. R., *Mixed Beta-gamma Exposures in Film Badge Dosimetry, AEC Rept. NYO-4565*, 1953) and of their individual dosage rates.

Choice of Film. The characteristic curves of Fig. 10-37a show that the density increment corresponding to a given exposure *ratio* increases with density. Therefore, within limits, the higher the density, the greater will be the accuracy of determination of the exposure associated with that density. Films suitable for radiation monitoring are available in a wide range of speeds (see Tables 10-4 and 10-5.) (See discussion of use of photographic films for monitoring on p. 10-75.) The film chosen for a particular application should have a speed such that the density corresponding to a significant exposure is well up off the "toe," or low-density portion, of the characteristic curve. Extremely high densities can be used with some films (see Tables 10-4 and 10-5). However, if the linear relation between net density and exposure is used in interpretation of monitoring films, the maximum density must be limited to a value found by experiment to lie on an acceptable approximation to the straight line.

Control films should be developed with each batch of monitoring films. These control films should be taken from the same batch as the monitoring films and should have been subjected to the same storage conditions but have not been intentionally exposed. Therefore, if the control films show any effect other than normal fog, it may be presumed that the same accidental exposure or adverse storage conditions have spoiled the other films also.

It is generally advisable to mount a **metal marker** over a monitoring film, that is, something that will cast a shadow to indicate that the density observed is due to penetrating radiation and not to some other cause. A strip of lead, 0.5 mm thick or more, is usually satisfactory. With the higher-energy radiations (200 kv or above), it is helpful to interpose 0.25 mm of copper between the lead and the film to reduce the intensifying action of the lead and thereby obtain a deeper shadow. Commercially available film badges usually incorporate some form of metallic marker whose shadow can be used for the above purpose.

At temperatures up to 100°F, the **keeping properties of unexposed film** are usually quite good, provided relative humidity does not exceed 50 per cent. The packages in which monitoring packets are supplied are often vaporproof. Packets so packed are not subjected to ambient humidity conditions until the vaporproof wrapping is opened to remove the individual packets. When, however, unprotected packets are exposed to higher humidity combined with the higher temperatures, deterioration is apt to be more rapid, particularly in the matter of increased fog. If monitoring films are to be exposed under such conditions, it is highly desirable that films to be used for calibration exposures be stored under similar conditions until the time of exposure, and that they be processed with the monitoring films.

Variation of Response with Temperature. The response of a film may vary with the temperature at which it is exposed, although the temperature at which it is stored before or after exposure has no appreciable effect. In the range from room temperature to 99°F, which might be encountered when a monitoring film is worn

Table 10-4. Approximate Film Sensitivities to Heavily Filtered 50-kv Radiation

Du Pont Dosimeter Films *

Film	Roentgens for net density of †					High reference	
	0.05	1.0	2.0	3.0	4.0	Exposure, roentgens †	Density
Type 502.....	0.005	0.10	0.19	0.70		1.1	3.13
Type 508.....	0.003	0.05	0.18			0.9	2.64
Type 510.....	0.017	0.55	1.5	2.5	3.4	4.4	5.00
Type 555.....	0.003	0.045	0.14	0.34	0.80	2.2	5.00
Type 834.....	0.16	2.7	9	55		90	3.40
Type 1290.....	0.68	20	70			700	2.90

Kodak Films ‡

Film	Roentgens for net density of ¶					High reference	
	0.05	1.0	2.0	3.0	4.0	Exposure, roentgens ¶	Density
Kodak Personal Monitoring Film, Type 2:							
Both emulsions.....	0.0013	0.032	0.075	0.14	0.29	0.68	5.0
Slow emulsion.....	0.094	1.9	4.9	12	46	100	4.3
Kodak Blue Brand Medical X-ray Film	0.0019	0.060	0.18			1.1	2.9
Kodak Industrial X-ray Film, Type AA...	0.0035	0.10	0.20	0.35	0.45	0.65	5.0
Kodak Fine Grain Positive Film.....	0.18	3.5	7.5	14	25	65	5.0

* Data supplied by E. I. du Pont de Nemours and Company, Photo Products Department.

† Development: 3 min at 68°F with intermittent agitation in a solution prepared from du Pont Liquid Concentrated X-ray Developer.

‡ Data supplied by Eastman Kodak Company.

¶ Development: 5 min in Kodak Rapid X-ray Developer or Kodak Liquid X-ray Developer and Replenisher at 68°F.

Table 10-5. Approximate Film Sensitivities to Radium Gamma Rays

Du Pont Dosimeter Films *

Film	Roentgens for net density of †					High reference	
	0.05	1.0	2.0	3.0	4.0	Exposure, roentgens †	Density
Type 502.....	0.07	1.6	5.6			40	2.85
Type 508.....	0.04	1.0	4.0			30	2.46
Type 510.....	1.0	16	30	45	56	68	5.00
Type 555.....	0.04	0.88	2.5	6.0	14	35	5.00
Type 834.....	3.0	35	130	500		700	3.10
Type 1290.....	9.0	190	700			3,000	2.70

Kodak Films ‡

Film	Roentgens for net density of ¶					High reference	
	0.05	1.0	2.0	3.0	4.0	Exposure, roentgens ¶	Density
Kodak Personal Monitoring Film, Type 1.....	0.013	0.29	0.61	1.0	1.5	2.4	5.0
Kodak Personal Monitoring Film, Type 2: Both emulsions.....	0.027	0.65	1.6	3.0	5.9	14	5.0
Slow emulsion.....	2.6	51	140	320	1,300	2,800	4.3
Kodak Blue Brand Medical X-ray Film.....	0.045	1.1	3.5			22	2.9
Kodak Industrial X-ray Film, Type AA.....	0.08	1.9	4.0	6.5	9.0	12	5.0
Kodak Fine Grain Positive Film....	6.0	80	170	300	600	1,500	5.0

* Data supplied by E. I. du Pont de Nemours and Company, Photo Products Department.

† Development: 3 min at 68°F with intermittent agitation in a solution prepared from du Pont Liquid Concentrated X-ray Developer.

‡ Data supplied by Eastman Kodak Company.

¶ Development: 5 min in Kodak Rapid X-ray Developer or Kodak Liquid X-ray Developer and Replenisher at 68°F.

either on the clothing or next to the skin, the dose indication was found, for a particular film, to be about 14 per cent higher for the higher temperature (Bass, H., Variation of Film Response with Temperature, *AEC Rept.* NYO-1522, 1950). In area monitoring with film, particularly outdoors, the range of ambient temperatures might be much greater. In going from 10 to 140°F, speeds of certain industrial and medical X-ray films to direct X-radiation have been found to increase by as much as 30 per cent, while others showed no variation in speed over the same range. In all cases, however, the sensitivity was found to be a linear function of temperature (Morgan, R. H., A Quantitative Study of the Effect of Temperature on Sensitivities of X-ray Screens and Films, *Radiology*, vol. 43, p. 256, September, 1944. Morrison, A., and K. J. Parry, Temperature and Intensifying Screen Effects in Radiography with Cobalt-60, *Non-Destructive Testing*, vol. 9, no. 1, p. 9, summer, 1950). For greatest accuracy in correlating exposures made at temperatures covering a wide range, the variation of response with temperature should be experimentally investigated for the particular film involved.

Latent-image Stability. Most films used for radiation monitoring in the range of the maximum permissible dose for humans are similar to medical and industrial X-ray films. With such films, generally, the **fading of the latent image** is very slight; that is, several days or even up to 2 weeks may elapse between exposure and processing without seriously affecting the developed density or the dosage measurement derived from it ("Radiography in Modern Industry," Eastman Kodak Co., Rochester, N.Y., 1957), provided that the exposed strips are kept at room temperature or below and that the relative humidity is below about 60 per cent. Therefore, in personnel monitoring, latent-image fading will lead to only very small errors.

However, with certain films, fading of the latent image can introduce serious errors in dosage measurement. For instance, with a very high resolution Lippmann-type film, useful for monitoring hard-gamma-ray doses of 10^3 to 10^4 r, storage for 2 weeks between exposure and processing can lead to a 60 per cent error in dosage measurement (McLaughlin, W. L., and M. Ehrlich, Film Dosimetry: How Much Fading Occurs? *Nucleonics*, vol. 12, no. 10, p. 34, October, 1954).

Two methods may be used to avoid difficulties from this cause. (1) Calibration films may be exposed to known radiation dosages at such a time that they will suffer about the same latent-image fading as the monitoring films. (2) The effects of the latent-image fading may be calculated from the formula which gives the density D_a at an arbitrary time a in terms of the developed density D_t at time t after exposure

$$\frac{D_a - D_t}{D_a} = 1 - e^{c(a^n - t^n)}$$

c and n are constants whose values must be determined by a separate latent-image-fading experiment for the particular film and development technique in question (McLaughlin, W. L., and M. Ehrlich, Film Dosimetry: How Much Fading Occurs? *Nucleonics*, vol. 12, no. 10, p. 34, October, 1954). This latter method must be used with caution, since the rate of fading of the latent image increases with increasing relative humidity. Therefore, the values of the constants c and n will depend upon the relative humidity of the air with which the film is in equilibrium and hence may depend upon the weather or the time of year.

Processing. Monitoring films are almost universally processed in developers used for medical and industrial X-ray films. The manufacturer's recommendations on details of processing, i.e., development time and temperature, and times of fixation and washing, should be followed. Developers are normally used as close to 68°F (20°C) as is possible in the particular installation, and development times range from 3 to 5 min.

The **speed** and the **contrast** (the steepness of the characteristic curve) of monitoring films can be controlled by the nature of the developer and the developing time. However, since the developers normally used in radiation monitoring are those which give the maximum speed and contrast, the speed and contrast of a particular film can be reduced only by the use of a developer other than the one recommended. This technique has been found useful in certain special problems.

By themselves, photographic films cannot give a measurement of the quantity of radiation. They must be suitably calibrated so that the densities observed on the monitoring films can be evaluated in terms of roentgens. The **degree of development**, and hence the density obtained, depends on the time of development, the temperature of the developer, its degree of exhaustion, and the amount of agitation. The effects of these factors, if unrecognized or unallowed for, may lead to serious positive or negative errors in photographic dosimetry. If **calibration films** can be developed with each batch of monitoring films, errors in processing will be minimized. Ordinarily, some films of the same emulsion number as the films worn by personnel are given a known exposure, or a series of exposures, expressed in roentgens, to primary radiation of suitable quality. These calibration films are processed with the monitoring films, and the exposures on the latter are evaluated by comparing their densities with those of the calibration films. If calibration films are developed with each batch of monitoring films, a high degree of constancy of the development conditions is not essential, although it is desirable to maintain recommended processing conditions.

Calibration films are usually necessary in any event to take account of the variation of sensitivity with radiation quality. If several calibration films, exposed to different amounts of the chosen radiation, are processed with each batch of monitoring films, their known dosages and measured densities can be used both to obtain the characteristic curve shape and its location along the log exposure axis, or to determine the slope of the net density vs. exposure curve. Alternatively, the characteristic curve shape can be determined by a series of exposures, of *known relative value*, to any convenient radiation; its location can be determined by a few films exposed to a properly chosen calibrating radiation.

However, if by the nature of the operation calibration films can be used only occasionally or not at all, extreme care must be exercised in maintaining constant all the developing conditions. **Errors in time of development** or in temperature can cause errors in dosage estimation of as much as plus or minus 30 per cent, and the magnitude of the error will be a function of the developed density (Corney, G. M., *Relation of Film Characteristics to X- and Gamma-Ray Monitoring*, *Nucleonics*, vol. 10, no. 11, p. 84, November, 1952). It should again be emphasized that errors from this cause can be entirely avoided by a proper use of calibration films.

In some large installations, where the processing conditions are under the strictest control, it has been found possible to reduce the numbers of calibration films used.

This is done by referring the densities of the monitoring films to a **"standard" calibration curve** made by averaging many previous calibration curves for the same film and precisely the same development conditions (Boyer, D. G., Film-badge Dosimetry: Development of a Standard Calibration Curve, *Nucleonics*, vol. 13, no. 11, p. 106, November, 1955. O'Brien, K., *et al.*, A System of Film Badge Interpretation, *Nucleonics*, vol. 14, no. 10, p. 58, October, 1956). Even under these conditions, however, it is advisable to process one or two films exposed to a known number of roentgens of a particular radiation. This will guard against gross errors such as might be caused by, for instance, the sudden failure of the timer or of the developer-temperature thermostat.

Processing Devices. Small numbers of monitoring films can conveniently be processed in open tanks on the hangers used for dental X-ray films. These accommodate up to 16 films each, and two hangers can be held in each hand for manual agitation to assure evenness of development. For larger numbers of films, processing racks of plastic or stainless steel (AISI 316 with 2 to 3 per cent molybdenum) accommodating up to several hundred films may be constructed. With such racks manual agitation of the racks may be difficult. However, agitation of the developer with pumps or stirrers should be viewed with caution. With any tank containing loaded film hangers, it is almost impossible to prevent steady-flow conditions. The uniform flow of developer along certain paths within the tank may cause more uneven development than no agitation at all.

Developer agitation by means of motor-driven paddles gives a high degree of turbulence and has been found suitable for the precise processing necessary in product control in the photographic industry (Jones, L. A., *et al.*, A Developing Machine for Sensitometric Work, *J. Soc. Motion Picture Engrs.*, vol. 28, p. 73, January, 1937). Such a device could be adapted to the processing of monitoring films. **"Gaseous burst" agitation**, used successfully in processing of color films, should also be applicable (Smibert, J. A., and M. O'Brien, Uniform Development by means of Intermittent Gaseous Burst Agitation, *Proc. Roy. Phot. Soc. Centenary Intern. Conf. Science Application of Photography*, Royal Photographic Society, London, 1955, p. 471. "Gaseous Burst Agitation in Processing," Pamphlet E-57, Eastman Kodak Co., Rochester, N.Y., 1956). In this method, large numbers of bubbles of nitrogen are periodically introduced into the bottom of the developer tank and produce a high degree of turbulence of the developer. The gas cannot be introduced continuously, however, because steady-flow conditions, leading to streaking, are likely to be set up. Other **developing mechanisms** have also been described (Freihofer, F., Continuous Processing of Dental-type Film, *Phot. Eng.*, vol. 2, no. 3, p. 203, 1951. Kardas, R. S., Small Sensitometric Developing Machine, *Phot. Eng.*, vol. 4, no. 3, p. 169, 1953. Magill, R. J., Simplified Film Processing for Radiation Dosimetry, *Nucleonics*, vol. 12, no. 8, p. 43, August, 1954).

In any event, **testing of a mechanical processing system** must be very carefully performed before the system is put into routine use. This can be done by processing a complete batch of films which have all been given the same exposure. The developed density should be such that it is on the steep portion of the characteristic curve—at least a density of 1.5 for most films. The films should be carefully identified as to their location in the processing machine. Density readings on the processed films will indicate whether or not the processing is sufficiently uniform,

and the densities can be referred to the characteristic curve to interpret any density differences found in terms of percentage error in dosage measurement. In addition, the individual films should be examined on a properly masked illuminator for evidence of streaking indicative of uneven development over the area of a single film.

The **materials for construction of processing apparatus** must be carefully chosen, since many common materials—for example, copper—have an exceedingly deleterious effect on photographic solutions [Muehler, L. E., and J. I. Crabtree, *Materials of Construction for Photographic Processing Equipment*, *PSA Journal (Phot. Sci. Tech.)*, vol. 19B, pp. 79, 92, May, August, 1953].

Commercially available **densitometers** are suitable for reading monitoring films. Visual instruments are cheaper, but the maximum density readable is about 3.0. Photoelectric densitometers are more rapid and convenient, and some can read densities as high as 5.0. Densities may be read on either type by experienced personnel to about plus or minus 0.01. Provided all films, including calibration and control films, of a certain batch are read on the same instrument, reproducibility of readings is much more important than absolute accuracy. If the density on an occasional double-coated monitoring film is too high to be read on the available densitometers, the emulsion may be removed from a small area on one side of the film, and from another area on the opposite side [Sherwood, H. F., *Methods for Eliminating the Image on One Side of Double-coated X-ray Film*, *PSA Journal (Phot. Sci. Tech.)*, vol. 19B, p. 151, November, 1953]. The densities are then read in both areas, and the values added to give the total density, a suitable correction being made for the density of the film base.

The **permanence of the developed image** on a monitoring film can be very high, provided fixation and washing have been carefully carried out (Crabtree, J. I., *et al.*, *Fixing and Washing for Permanence*, *J. Phot. Soc. Amer.*, vol. 9, p. 115, March, 1943; vol. 9, p. 162, April, 1943). In general, **archival quality** (i.e., unchanged indefinitely under good storage conditions) will be attained for X-ray film when the hypo content of the processed film has been reduced to between 0.01 and 0.03 mg of anhydrous sodium thiosulfate per square inch of film. **Permanence for commercial use** (remain unchanged for several years under normal storage conditions) will be attained at somewhat over 0.10 mg/sq in. ("Instructions for Use of the Kodak Hypo Estimator," Eastman Kodak Co., Rochester, N.Y., 1955). The residual hypo in a processed film may be estimated ("Instructions for Use of the Kodak Hypo Estimator," Eastman Kodak Co., Rochester, N.Y., 1955) or determined (American Standard Method for Determining the Thiosulfate Content of Processed Photographic Film, PH 4.8-1953. American Standards Association, New York, 1953. Crabtree, J. I., *et al.*, *A Review of Hypo Testing Methods*, *J. Soc. Motion Picture Engrs.*, vol. 42, p. 34, January, 1944). In addition, methods are available for determining the **stability of already processed materials**, provided some samples may be sacrificed for a destructive test (American Standard Method for Indicating the Stability of the Images of Processed Black-and-white Films, Plates, and Papers, PH 4.12-1954, American Standards Association, New York, 1954). In one instance, it has been found that the density stability of developed monitoring films has remained essentially unimpaired for at least 3 years of storage under what were probably average conditions for northern United States (O'Brien,

K., and L. Solon, Effect of Age on the Reproducibility of Film Badge Density Readings, *AEC Rept.* NYO-4576, 1954).

MONITORING OF X- AND GAMMA RAYS

The **X-ray spectral sensitivity curve** for a typical X-ray film is shown in Fig. 10-38 (Wilsey, R. B., The Use of Photographic Films for Monitoring Stray X-rays and Gamma Rays, *Radiology*, vol. 56, p. 229, February, 1951), in which is plotted the number of roentgens required to produce a density of 1.0 as a function of radiation energy. The same form of curve has been found experimentally for other films

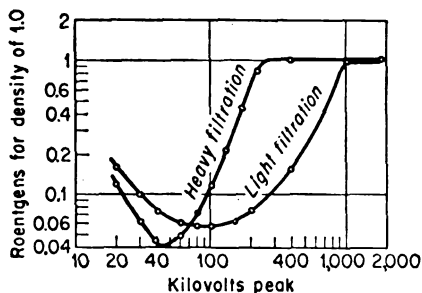


FIG. 10-38. Roentgens for a density of 1.0 as a function of X-ray tube voltage, for light and heavy filtration, on Kodak Blue Brand Medical X-ray Film. (After Wilsey, *Radiology*, vol. 56, p. 229, 1951.)

used in monitoring (e.g., Pardue, L. A., *et al.*, Photographic Film as a Pocket Radiation Dosimeter, *AEC Rept.* MDDC-1065, 1948. Deal, L. J., *et al.*, Roentgen-ray Calibration of Photographic Film Exposure Meter, *Am. J. Roentgenol.*, vol. 59, p. 731, May, 1948. Tochilin, E., *et al.*, A Calibrated Roentgen-ray Film Badge Dosimeter, *Am. J. Roentgenol.*, vol. 64, p. 475, September, 1950. Storm, E., The Response of Film to X-radiation of Energy up to 10 Mev., *Los Alamos Sci. Lab. Rept.* LA-1220, 1951), although the exact shape of the curve depends upon the particular radiations, or radiation bands in the case of filtered X-rays, used in the experiment. Details of the curve, including the ratio of maximum to minimum sensitivity, also vary from film to film, as can be seen by comparing the ratios of corresponding values in Tables 10-4 and 10-5. All investigations agreed, however, in showing a maximum sensitivity for the radiation quality corresponding to lightly filtered 80- to 100-kv, or heavily filtered 50-kv, X-rays (see Table 10-4), and a broad, flat minimum of sensitivity for radiations harder than heavily filtered 250-kv X-rays and including the gamma rays of radium and Cobalt-60 (see Table 10-5).

Evidence exists that the sensitivity of photographic films increases somewhat for radiations above about 2 Mev (Storm, E., The Response of Film to X-radiation of Energy up to 10 Mev, *Los Alamos Sci. Lab. Rept.* LA-1220, 1951), but the difficulties of ionization measurements in this range introduce experimental difficulties into the measurements.

Consideration of curves of the type of Fig. 10-38 shows that large errors (up to a factor of 25 in either direction) in dosage measurements can occur if proper cognizance is not taken of the spectral sensitivity of the monitoring film. To avoid difficulties on this score, calibration films should be exposed to known dosages of carefully chosen radiations (see p. 10-63). These calibration films are then processed along with the monitoring films and control films, and the densities of the monitoring films are interpreted in roentgens by comparing them with the densities of the calibration films. This, in turn, means that either a reliable ionization chamber, a

calibrated gamma-ray source, or both, must be available for determining with acceptable accuracy the dosages delivered to the calibrating films.

In making the **film-calibration exposures**, sources of stray radiations should be reduced to a minimum. Thus, with X-rays the primary beam should be diaphragmed to a small size, but sufficient to cover the ionization chamber or the films; tube filters and diaphragms are sources of stray radiation and should be mounted at or near the X-ray tube portal. With any radiation, the ionization chamber, or films, should be spaced well away, preferably 1 m or more, from any object, such as a room wall, struck by the primary beam. Secondary radiation from the exposed portion of a lead wall can be greatly reduced by covering the wall with sheet iron or steel overlaid with a $\frac{1}{4}$ -mm thickness of aluminum. Exposing the chamber and the films at the same location will ensure that both receive the same radiation intensity, but if different distances must be used because of the difference in sensitivity between chamber and film, the precautions given above to limit stray radiation will minimize any deviations from the inverse-square law.

In order to arrive at photographic measurements of radiation dosage which are not unduly pessimistic or dangerously optimistic, one of three conditions must be met: (1) the quality of the radiation to which the monitoring films have been exposed must be known; (2) the spectral response of the film must be made flat over

Table 10-6. Calibrating Radiations *

Application	Remarks	Calibrating radiation
Medical diagnostic radiography	Max voltage 100 kvp	Voltage most commonly used in X-ray examinations
Medical diagnostic radiography	Max voltage 125 kvp	Direct 125-kvp X-rays
Medical diagnostic radiography	All work with very narrow beams	Heavily filtered primary radiation of average voltage used
Medical therapy	Max voltage 250 kv	Heavily filtered primary radiation; Ra or Co-60 gamma rays
Medical therapy	Max voltage 1,000 kv	Primary radiation; Ra or Co-60 gamma rays
Industrial radiography	Max voltage 250 kv	Heavily filtered primary
Industrial radiography	Max voltage 1,000 kv	Primary radiation †
Industrial radiography	Co-60	Primary radiation †

* Wilsey, R. B., *et al.*, Experiments in the Photographic Monitoring of Stray X-rays. I. General Considerations. The Choice of Film Calibrating Radiations in Roentgen Therapy at 220 kvp and 1,000 kvp, *Radiology*, vol. 66, p. 408, March, 1956.

Ibid., II. The Characteristics of the Stray Radiations and the Choice of Film Calibrating Radiations in Diagnostic Radiology, p. 418.

Wilsey, R. B., *et al.*, The Photographic Monitoring of Stray X-rays in the Radiography of Metals, *Nondestructive Testing*, vol. 14, no. 2, p. 18, March-April, 1956.

† Monitoring films should be partly covered with 1 mm aluminum. Any great density difference between covered and uncovered sections indicates an exposure to electrons and should be investigated.

the radiation-quality range of interest; or (3) it must be possible to tell from the processed film the approximate radiation qualities comprising the exposure. If none of the above three conditions can be attained in a particular instance, it is still possible to obtain a measure of the maximum probable dose received by a film, by assuming that all the exposure was to hard radiation (e.g., radium gamma rays) for which the film has minimum sensitivity. This procedure will not lead to under-valuation of the radiation but obviously may result in an alarming or inconvenient overevaluation of soft radiation.

The simplest situation exists when the approximate radiation quality reaching monitoring films is known. The calibration films are then exposed to a radiation quality which will cause that radiation to which monitoring and calibration films are least sensitive (e.g., the primary radiation) to be evaluated correctly. Other radiations (e.g., the secondary radiation), then, will be overvalued—in the direction of increased safety. For instance, in monitoring the personnel of a medical diagnostic department, it might be known that the maximum X-ray tube voltage was 125 kv and that the personnel might be exposed to direct radiation, direct radiation filtered through a protective barrier, and radiation scattered from the patient and from structures in the direct beam. In this particular instance, exposure of calibration film to direct 125-kv X-radiation would result in the best evaluation of the radiation received by the personnel. Table 10-6 gives the **calibrating radiations** suitable for a number of applications of X- and gamma rays.

Filters to Reduce Energy Dependence. A metallic filter will have a preferential absorption for softer radiation. Thus, a layer of metal over a film will transmit a higher percentage of those radiations to which the film is least sensitive than those to which it is more sensitive. By this means, the response of a film-filter combina-

tion may be made effectively uniform over a wider range of wavelengths than that of a film alone.

Such filters are normally easy to use, since monitoring films, for convenience, are usually worn in a light metal or plastic badge which may be affixed to the clothing or worn on the wrist. Filters are often incorporated in these badges, which are available commercially (Commercial Film-badge Services, *Nucleonics*, vol. 13, no. 2, p. 86, February, 1955).

Probably the **commonest filter in current use** is about 1 mm of cadmium over a portion of the film, front and back (Wilsey, R. B., *The Use of Photo-*

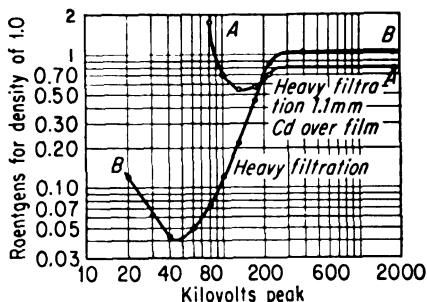


FIG. 10-39. Sensitivity of a typical monitoring film to heavily filtered X-rays. Curve A, with 1.1 mm cadmium over film; curve B, without cadmium. (After Wilsey, *Radiology*, vol. 56, p. 229, 1951.)

graphic Films for Monitoring Stray X-rays and Gamma Rays, *Radiology*, vol. 56, p. 229, February, 1951. Pardue, L. A., *et al.*, Photographic Film as a Pocket Radiation Dosimeter, *AEC Rept.* MDDC-1065, 1948. Deal, L. J., *et al.*, Roentgen-ray Calibration of Photographic Film Exposure Meter, *Am. J. Roentgenol.*, vol. 59, p. 731, May, 1948). Figure 10-39 shows the effect of such a filter on the spectral-

sensitivity curve of the film of Fig. 10-38. This particular filter-film combination will evaluate to within about ± 20 per cent all perpendicularly incident radiations harder than about 0.1 Mev (effective). Some other filters either proposed or in use are tabulated in Table 10-7.

Table 10-7. Typical Filters for Monitoring Films

Filter	Range and approx accuracy within range	Ref
1.0 mm Cd.....	See text	
0.5 mm Pb.....	0.2 Mev _{eff} to radium gamma rays; $\pm 20\%$	1
1.07 mm Sn + 0.03 mm Pb (superposed)	0.12 Mev _{eff} to 11 Mev betatron radiation; $\pm 20\%$	2
1.1 mm Sn.....	200 kv X-rays filtered to give HVL = 1.5 mm Cu to radium gamma rays; $\pm 10\%$ for 45° incidence, $\pm 25\%$ for normal incidence	3

1. Storm, E., and E. Bemis, A Method for Monitoring Using Eastman Type K Film, *Los Alamos Sci. Lab. Rept. LA-1107*, 1950.

2. Ehrlich, M., and S. H. Fitch, Photographic X- and Gamma-ray Dosimetry, *Nucleonics*, vol. 9, no. 3, p. 5, September, 1951; no. 6, p. 12, December, 1951.

3. Stephenson, S. K., Filters for Health Film Holders, *Brit. J. Radiol.*, vol. 26, p. 380, July, 1953.

The values for most filters are given for perpendicular incidence of radiation. For **radiations incident upon the filter at a lower angle**, the effective filter thickness is greater. This will lead to an underestimation of the radiation received by the badge. The underestimation is not serious, from a personnel-monitoring standpoint, for very penetrating radiations (e.g., 1 Mev) but can be very great for radiation as hard as 250-kv unfiltered X-rays (Ehrlich, M., "Photographic Dosimetry of X- and Gamma Rays," NBS Handbook 57, GPO, 1954). However, dangerous errors from this cause can be avoided by one of the methods described below.

Film badges usually have, in addition to a filter, an "open-window" section, in which the film packet is surrounded only by a small mass of low-atomic-number material. Such a badge can be used in the **monitoring of a mixture of hard and soft radiation**. Only the hard radiations will appreciably affect the area under the filter. The "window" area will be affected by both hard and soft radiation, but much more strongly by the latter. Two calibration curves are prepared, one for hard radiation and the filter-film combination, and the other for soft radiation on the film alone. The densities on the two areas of the monitoring film are referred to their individual calibration curves, and the resulting estimates of exposure in roentgens are added to give the total dose.

Estimation of Radiation Quality from Monitoring Film. If monitoring films are exposed to a wide and unknown range of X- or gamma-ray energies, conservative application of the above methods of interpretation may sometimes lead to over-estimations of the radiations received. Although such errors are in the direction of

increased safety for personnel, it may be desirable, where overvaluation is potentially severe, to have a more realistic estimate of the exposure. This in turn implies a means of determining the quality of the exposing radiation from the monitoring film itself.

A number of methods are available and have been used in practice. Only the general principles will be described here; the original references should be consulted for details since the advantages, limitations, and range of applicability of any method must be clearly suited to the particular monitoring problem in hand.

Single-filter Methods. Examination of the curves of Fig. 10-38 shows that the ratio between the exposure received by a film under a filter and that received through an "open window" is characteristic of the effective energy of the exposing radiation, at least in that range where film response is not essentially uniform. A method has been described involving the use of a 0.5-mm lead filter over part of the film. The densities in the two areas are interpreted as relative exposures by reference to the characteristic curve of the film. The ratio of these two relative exposures then determines uniquely the quality of the exposing radiation in a range from about 0.05 to 2.0 Mev_{eff} (Storm, E., *The Response of Film to X-radiation of Energy Up to 10 Mev*, *Los Alamos Sci. Lab. Rept.* LA-1220, 1951. Storm, E., and E. Bemis, *A Method for Monitoring Using Eastman Type K Film*, *Los Alamos Sci. Lab. Rept.* LA-1107, 1950). This method would require that both densities be high enough to be read accurately; otherwise a large error in their ratio is possible.

Multiple-filter Methods. Multiple filters may take the form of **stepped wedges** (echelons) of a single material or several individual filters differing in thickness and composition.

In the case of stepped wedges, the densities on the monitoring film under the several steps are interpreted as log relative intensities ($\log I$) from a characteristic curve of the film, and plotted against step thickness x . The slope, or effective slope, of this curve gives a value of $d \log I / dx$ which may be substituted into the differentiated logarithmic form of the exponential-absorption law:

$$\frac{d \log I}{dx} = - \frac{1}{2.303} \frac{\mu}{\rho} \rho$$

(ρ = density of filter material) to give a value of μ/ρ , the effective mass-absorption coefficient. From a table of mass-absorption coefficient vs. wavelength, the effective wavelength may be found.

Alternatively, a **filter array**, i.e., a number of different filters of fairly widely spaced atomic number, may be incorporated into the film badge. The relative absorptions of these filters will depend upon the wavelength of the exposing radiation, as will the developed densities beneath them. The density differences beneath pairs of filters will then be characteristic of the effective wavelengths of the exposing radiation. If all densities lie on an acceptably straight portion of the net density vs. exposure curve, the effective wavelength may be determined by referring the density ratios to a previously experimentally determined table of density ratio vs. radiation quality (or effective wavelength). Several multiple-filter methods are briefly described in Table 10-8. Equipment for multiple-filter methods is available com-

mercially, either as prepared badges or as radiation-monitoring services (Commercial Film-badge Services, *Nucleonics*, vol. 13, no. 2, p. 86, February, 1955).

Table 10-8. Multiple Filters for Estimation of Radiation Quality

Filters	Remarks	Ref.
"Open window," 1.1 mm Al, 0.25 mm steel + 0.76 Al, 1.1 mm Cd	Demonstrated useful for quality determination in medical diagnostic range, and to distinguish these radiations from radium gamma rays; probably useful for intervening radiation qualities	1
"Open window," 0.5 mm Ag, 0.5 mm Ag + 0.5 mm Pb	Visual discrimination between 85-kv scattered X-rays, 250-kv scattered X-rays, and radium gamma rays. Density measurements permit estimation of radiation quality	2
"Open window," 0.5 mm Al, 0.5 mm Al + 0.2 mm Cu, 0.5 mm Al + 0.6 mm Cu	17.5 kV_{eff} to 130 kV_{eff} (unfiltered 30 kvp to moderately filtered 250 kvp X-rays)	3
"Open window," 5 mm brass, 5 mm Pb	0.02 to 2 Mev_{eff}	4
"Open window," 0.2 mm Cu, 0.5 mm Cu, 1.0 mm Cu, 1.0 mm Pb	Visual discrimination between radium gamma rays and therapeutic X-rays. Quality of X-radiation can be determined	5

1. Hunter, F. T., *et al.*, The Protection of Personnel Engaged in Roentgenology and Radiology, *New Engl. J. Med.*, vol. 241, p. 79, July 21, 1949.
2. Baker, R., and L. B. Silverman, An Improved Film Badge Method for the Accurate Determination of Personnel Exposures, *AEC Rept. UCLA-53*, 1950.
3. Tochilin, E., *et al.*, A Calibrated Roentgen-ray Film Badge Dosimeter, *Am. J. Roentgenol.*, vol. 64, p. 475, September, 1950.
4. Storm, E., The Response of Film to X-radiation of Energy up to 10 Mev, *Los Alamos Sci. Lab. Rept.*, LA-1220, 1951.
5. Madsen, C. B., Problems in Radiation Monitoring with Film Badges, *Acta Radiol.*, vol. 37, p. 284, March-April, 1952.

Intensifying-foil Methods. Metals exposed to X-rays or gamma rays give off electrons, the energy and intensity of which are a function of radiation quality (Seemann, H. E., Some Physical and Radiographic Properties of Metallic Intensifying Screens, *J. Appl. Phys.*, vol. 8, p. 836, December, 1937. Corney, G. M., unpublished data). If a metal foil is placed in contact with a film, these electrons will intensify the photographic action of the radiation to a degree which is also dependent upon the radiation quality. By comparing the density beneath a metal foil with that of an uncovered area of film, an estimate of radiation quality may be obtained. Intensifying-foil methods have been combined with filter methods. In addition, step wedges of thin low-atomic-number materials have been incorporated between metal foil and film to analyze the electron radiation from the foil, in much the same way as X- and gamma radiation may be analyzed by a metal step wedge, in order to obtain an estimate of radiation quality. Two intensifying-foil methods are described in Table 10-9.

Table 10-9. Metallic Intensifying Foils for Estimation of Radiation Quality

Foils	Remarks	Ref.
"Open window," Pb, Al, Sn	Electron-absorbing filters between foils and film. Visual examination can indicate radiation quality if possibilities are limited. Density measurements permit estimation of unknown quality	1
Pb behind film	Electron-absorbing filters between foil and film to allow estimate of radiation quality	2

1. Spiegler, G., A New Method of Use of the Photographic Film as Quality Indicator and Dose Meter for X- and Gamma Rays, *Phot. J.*, vol. 90B, p. 166, 1950.

2. Soole, B. W., Photographic Badges for the Estimation of X and Gamma Radiation, *Brit. J. Radiol.*, vol. 24, p. 450, August, 1956.

Effect of Body Backscatter. Film badges and pocket dosimeters are usually calibrated in free air but under actual field conditions are worn on the person. It has been shown that these differences between calibration and field conditions make no practical difference in dose calibration in the measurement of hard gamma radiation (Solon, L. R., and H. Blatz, Effect of Body Backscatter in Gamma-ray Personnel Dosimetry, *Nucleonics*, vol. 13, no. 4, p. 62, April, 1955).

Electronic Equilibrium. From the standpoint of radiation protection, a dosimeter should register a dose equal to that received by the critical organs some distance below the skin. When a beam of very high energy photons strikes a block of material, however, the secondary electrons that are projected mainly in the forward direction do not build up to equilibrium with the photon intensity until a depth approximately equal to the average electron range is reached. By equilibrium is meant the condition in which as many electrons are *stopped* in a slice of the material as *originate* there (neglecting the small exponential decrease in photon intensity). This means that, as long as the material surrounding a dosimeter is thicker than the range of the secondary electrons, the secondary-electron density as measured by the dosimeter is proportional to the photon intensity at some point within the dosimeter wall.

In the case of photographic dosimeters, the condition of electronic equilibrium is comparatively well met up to about 0.3 Mev by film in its conventional paper wrapper. At higher energies, difficulties arise from the fact that the range of secondary electrons becomes greater than the thickness of the film plus paper wrapper. The secondary electrons, therefore, are not completely stopped in the emulsion, whereas these electrons would be absorbed in thick layers of living tissue. Thus, the photographic effect ceases to be a measure of deep-tissue dose for these energies, unless the film packet is surrounded with an additional layer of tissue-equivalent material whose thickness depends upon the energy of the photons measured. The thickness of the electronic equilibrium layer for Cobalt-60 gamma radiation is about 3 mm of a low-atomic-number material such as Bakelite plastic. For X-rays generated at 11 Mev the corresponding equilibrium layer is approximately 2 cm of Lucite

plastic. In the particular case of Cobalt-60 gamma rays, a film in an electronic-equilibrium layer of Bakelite plastic will indicate a dose about 20 per cent higher than will a bare film exposed under the same conditions (Ehrlich, M., "Photographic Dosimetry of X- and Gamma Rays," NBS Handbook 57, GPO, 1954). If personnel films are worn in an "equilibrium-layer" badge, the calibration films should be exposed in the same way.

BETA-RAY MONITORING

The **beta-ray sensitivity curves** for a typical double-coated X-ray film are shown in Fig. 10-40 (Dudley, R. A., *Photographic Detection and Dosimetry of Beta Rays*, *Nucleonics*, vol. 12, no. 5, p. 24, May, 1954), in which are plotted the number of electrons required to produce a density of 0.3 as a function of electron energy. These sensitivities are for electrons directly incident upon the bare film. The solid curves of Fig. 10-40 are for normally incident monoenergetic electrons. The "valley" between the maxima for the double-coated film represents the absorption of the film base, and the abrupt rise to the right of the "valley" represents the sensitivity of the back emulsion to electrons energetic enough to penetrate the front emulsion and the base. Film sensitivity is essentially uniform to monoenergetic electrons above about 0.5 Mev (Dudley, R. A., *Photographic Detection and Dosimetry of Beta Rays*, *Nucleonics*, vol. 12, no. 5, p. 24, May, 1954. Fleemann, J., and F. S. Frantz, *Film Dosimetry of Electrons in the Energy Range 0.5 to 1.4 Million Electron Volts*, *J. Research NBS*, vol. 48, p. 117, February, 1952. Jetter, E. S., and H. Blatz, *Film Measurement of Beta Radiation Dose*, *Nucleonics*, vol. 10, no. 10, p. 43, October, 1952).

The dotted curve of Fig. 10-40 is for **diffusely incident monoenergetic electrons** and shows that, for diffuse incidence, the effect of electron energy is less marked than for normal incidence.

The sensitivity of a bare film to the **continuous beta-ray spectra** emitted by radio isotopes is independent of beta-ray energy for continuous spectra having a maximum energy of 1.8 Mev or more; it is uniform to within plus or minus 10 per cent for beta-ray spectra above about 1.2 Mev maximum energy (Tochilin, E., and R. Golden, *Film Measurement of Beta-ray Depth Dose*, *Nucleonics*, vol. 11, no. 8, p. 26, August, 1953).

The above values are for bare films. However, films used for monitoring are always enclosed in an opaque paper wrapper having a thickness of about 30 mg/cm². This is approximately the range for 0.15-Mev electrons (Glendenin, L. E., *Determination of the Energy of Beta Particles and Photons by Absorption*, *Nucleonics*,

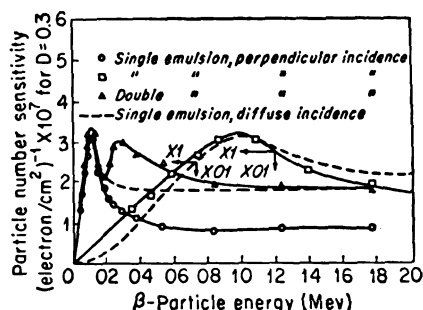


FIG. 10-40. Sensitivity of a typical X-ray film to monoenergetic electrons. Solid curves—normal incidence; dashed curves—diffuse incidence. (From Dudley, *Nucleonics*, vol. 12, no. 5, p. 27, May, 1954.)

vol. 2, no. 1, p. 12, January, 1948). Thus very little beta radiation below that energy will reach the wrapped film, with the corresponding deformation of the energy-sensitivity curve. However, the **sensitivity of conventionally wrapped films** is independent of maximum energy of beta radiation above about 2.5 Mev, and uniform to within plus or minus 10 per cent above a maximum energy of about 1 Mev (Storm, E., Response of Sensitive 552 Du Pont Film to Beta Radiation, *Los Alamos Sci. Lab. Rept. LA-1284*, 1951).

Normal beta-ray monitoring usually involves diffusely incident electrons of the continuous spectrum of beta-active isotopes. In this case, calibration exposures are most often made by putting films in contact with a massive beta emitter whose output has been measured by an extrapolation ionization chamber. In the case of uranium beta rays, the calibration source is most conveniently a thick piece of metallic uranium. Calibration sources of other isotopes are commercially available or may be made by dispersing the isotope in a plaster of paris disk to provide an "infinitely thick" source.

Where exposures to personnel are to the beta rays from two isotopes differing in maximum energy, the calibration exposures can be made with that isotope to whose radiation the film is least sensitive. This procedure will result in correct evaluation of one of the radiations and overvaluation of the other, an error in the direction of increased safety.

If, in the above circumstances, it is desired to avoid overvaluation of the radiation received, a value for the **effective maximum energy of the mixed beta radiation** may be found. If part of a film packet is covered with a low-atomic-number absorber (about 30 mg/cm²) the exposure difference in the two parts of the film will be a function of effective beta-ray energy. At low energies, the absorption of the filter is large but decreases rapidly with increasing beta-ray energy until it is almost zero at about 3.5 Mev. The ratio of relative exposures or, if the densities lie on the acceptably straight portion of the net density vs. exposure curve, the ratio of net densities on the two parts of the film will characterize the effective energy of the mixture of beta-ray spectra to which the films have been exposed. By using a number of isotopes of differing beta-ray energy, a curve is drawn relating this exposure ratio with beta-ray energy, for the particular film involved. Once this relationship is known, the relationship between the densities of the two parts of the monitoring films indicates the effective energy to which they have been exposed, and the densities can be evaluated on the appropriate calibration curve (Storm, E., Response of Sensitive 552 Du Pont Film to Beta Radiation, *Los Alamos Sci. Lab. Rept. LA-1284*, 1951). This method, of course, requires that calibration exposures made with several qualities of beta radiation be processed with the monitoring films.

NEUTRON MONITORING

Any type of **neutron monitoring** presents more difficulties of performance and interpretation than does the monitoring of other radiations. Most of the difficulties in photographic monitoring stem from the fact that neutrons, being nonionizing, do not directly affect the photographic film. However, neutrons may be made to produce photographically effective radiations which can be evaluated. Two general methods are used. Neutrons may be made to produce beta or gamma activities,

resulting in a film density which is read on a densitometer. Alternatively, particle tracks may be produced in "nuclear-track" emulsions, and the track population may be counted under the microscope and correlated with neutron dose. **Densitometric evaluation** is quicker and more convenient but has a low discrimination against other radiations which may be present. **Track counting**, while more laborious, is more accurate and often allows discrimination among various radiations simultaneously incident on the film. Fast neutrons produce knock-on protons in hydrogen-containing materials and slow neutrons induce nuclear reactions with the resulting emission of fast particles. Most neutron monitoring is done in this manner.

The counting of tracks, being both difficult and tedious, is the source of several technical and psychological difficulties. The longer tracks are more easily seen and more likely to be counted than are the shorter. Tracks only three grains long are difficult to observe; isolated grains or pairs of grains due to particle exposure cannot be distinguished from fog grains. Since, in some cases, track length is a function of neutron energy, the above effects contribute to the energy dependence of the film badge.

Calibration exposures should be made to neutrons of about the energy to which personnel will be exposed because of the dependence of track length, and hence of track visibility, upon neutron energy. Usually exposures are made to several known levels of neutron flux, preferably at the start of the monitoring period. Calibration films are often introduced randomly into the monitoring films in order to check on the accuracy of the counting procedures.

In the counting of tracks, it has been found necessary to select arbitrarily the microscopic fields to be viewed, since the natural tendency of an observer is to choose those fields containing the most tracks (Watson, E. C., *Fast Neutron Monitoring of Personnel*, HW 21552, General Electric Co., Nucleonics Division, Richland, Wash., 1951). Instead of counting the tracks in a number of discrete fields, the same total area of the film may be scanned with an increase in accuracy of track count (Cheka, J. S., *Recent Developments in Film Monitoring of Fast Neutrons*, *Nucleonics*, vol. 12, no. 6, p. 40, June, 1954). In any event, the careful examination of a single monitoring film may take from 30 to 60 min (Watson, E. C., *Fast Neutron Monitoring of Personnel*, HW 21552, General Electric Co., Nucleonics Division, Richland, Wash., 1951. Titterton, E. W., and M. E. Hall, *Neutron Dose Determination by the Photographic Plate Method*, *Brit. J. Radiol.*, vol. 23, p. 465, August, 1950).

The tracks of ionizing particles are subject to **latent-image fading** between exposure and development, and the rate of fading is dependent upon relative humidity. A desiccated sample (at 24°C) of a particular neutron-monitoring film was found to lose 12 per cent of its tracks in about 60 days after exposure, one kept at 24°C and 50 per cent relative humidity lost the same number of tracks in 16 days, while a film kept under "unmodified ambient conditions" suffered the same loss in only 5 days (Cheka, J. S., *Recent Developments in Film Monitoring of Fast Neutrons*, *Nucleonics*, vol. 12, no. 6, p. 40, June, 1954). Further, since all grains produced by a given ionizing radiation have the same possibility of fading, latent-image fading preferentially affects the shorter tracks, introducing another artificial energy dependence into the film. Calibration films should be exposed at the start of a monitoring period and stored until the monitoring films are developed. Thus, latent-

image fading will be more severe on the calibration films, and any errors will be in the direction of increased safety.

Despite these drawbacks, films are used for neutron monitoring because they are light, cheap, and easy to wear and provide a permanent record of the exposure, a valuable feature if the maximum permissible dose should sometime be revised downward.

Fast Neutrons. The original personnel-monitoring method for fast neutrons relied upon the body of the wearer to moderate some of the flux to thermal neutrons, which in turn underwent an (n,γ) reaction in a cadmium shield over part of a conventional gamma-ray monitoring film. The density due to the induced gamma rays was correlated with fast-neutron flux by exposing calibration films placed on a water or paraffin phantom which simulated a human body (Dessauer, G., and E. Lennox, *Photographic Neutron Dosimetry to Date*, *AEC Rept.* AECD-2278, 1948. Dessauer, G., and J. Rouvina, *Photographic Neutron Dosimetry II*, *AEC Rept.* AECD-1973, 1948). Because of a decrease in the maximum permissible neutron exposure, this method is no longer considered of adequate sensitivity for personnel monitoring (Rausa, G. J., *Comparison of the Photographic Effects Produced by Cadmium and Rhodium after Neutron Bombardment with Reference to Personnel Monitoring*, UR-253, University of Rochester Atomic Energy Project, Rochester, N.Y., 1953).

Most current methods of fast-neutron monitoring depend upon the tracks of recoil protons produced when fast neutrons interact with a hydrogen atom in the gelatin of the emulsion or in the film base. In a typical neutron-monitoring film one track is produced for about 1,500 incident neutrons, if the film is developed immediately after exposure. Two weeks of maximum permissible exposure rate (50 neutrons/cm²/sec) would represent about 75 tracks per 40 microscopic fields counted. Slow neutrons will also produce proton tracks from a reaction with the nitrogen in the gelatin of the emulsion, but the number of slow neutrons per observed track is about 25 to 30 times the value for fast neutrons (Watson, E. C., *Fast Neutron Monitoring of Personnel*, HW 21552, General Electric Co., Nucleonics Division, Richland, Wash., 1951). The resulting overvaluation of fast-neutron flux is not serious, considering the statistical nature of the observation.

A film may be made to give a response (in terms of tracks per incident neutron) parallel, over a wide energy range, to the neutron dose (in terms of reps per incident neutron) by surrounding it with the proper hydrogenous and nonhydrogenous layers. In practice paper and aluminum, respectively, are used, and most of the layers are packed within a commercially available film packet. This packet, in turn, may be used in a special aluminum and cellulose badge. The combination gives a response parallel to the neutron dose over a neutron-energy range of 0.25 to 14 Mev. The special packet in a conventional badge is usable in the energy range 0.25 to 10 Mev (Cheka, J. S., *Recent Developments in Film Monitoring of Fast Neutrons*, *Nucleonics*, vol. 12, no. 6, p. 40, June, 1954).

Thermal neutrons may be monitored both by a reading of density and by a track-counting method.

Thermal neutrons will induce beta-ray activities with short half-lives in silver, indium, and rhodium, and gamma activity in cadmium. A layer of these metals over a conventional X- or gamma-ray monitoring film will produce a density related to the neutron flux. Both **cadmium** and **rhodium converter foils** have been

investigated. Cadmium was found to give a lower developed density than rhodium for the same flux (Rausa, G. J., Comparison of the Photographic Effects Produced by Cadmium and Rhodium after Neutron Bombardment with Reference to Personnel Monitoring, UR-253, University of Rochester Atomic Energy Project, Rochester, N.Y., 1953). However, either might be used in personnel monitoring depending upon the sensitivity required.

Track-counting methods depend upon the interaction of thermal neutrons with the nitrogen of the emulsion gelatin, resulting in proton tracks, or with lithium or boron, with which the emulsion may be loaded, giving tritium and alpha tracks. Loading with lithium may increase the sensitivity of the film to thermal neutrons by a factor of 15, and with boron by a factor of about 200 (Titterton, E. W., and M. E. Hall, Neutron Dose Determination by the Photographic Plate Method, *Brit. J. Radiol.*, vol. 23, p. 465, August, 1950).

Mixed neutrons may be monitored by a track-counting method using a film partially shielded with cadmium or rhodium. Either metal has a high cross section for thermal neutrons, the absorption of which results in gamma emission. Any tracks, therefore, in the shielded section will be due to knock-on protons produced by fast neutrons and will indicate fast-neutron exposure. In the unshielded part, the tracks will be due to both recoil protons from fast neutrons and protons from the $N(n,p)$ reaction of thermal neutrons. The track population here, then, will be a function of total neutron exposure from which the thermal-neutron exposure can be found by difference (Cheka, J. S., Neutron Monitoring by Means of "Special Fine-grain Alpha-emulsion" Film, *AEC Rept. MDDC-890*, 1947).

Alternatively, mixed-neutron exposures can be monitored by having the personnel wear separate "fast" and "thermal" badges.

MONITORING OF MIXED RADIATIONS

Beta and Gamma Radiation. The photographic effect of simultaneously or consecutively incident beta or gamma rays is independent of the order and/or rate at which the radiations are delivered (Solon, L. R., Mixed Beta-gamma Exposures in Film Badge Dosimetry, *AEC Rept. NYO-4565*, 1953). They may be measured independently by covering part of the film with a filter which absorbs essentially all the beta radiation but which has little effect on the X- or gamma radiation.

The density under the filter is due to X- or gamma radiation alone, that under the "open window" to beta plus gamma radiation (Storm, E., Response of Sensitive 552 Du Pont Film to Beta Radiation, *Los Alamos Sci. Lab. Rept. LA-1284*, 1951. Solon, L. R., Mixed Beta-gamma Exposures in Film Badge Dosimetry, *AEC Rept. NYO-4565*, 1953. O'Brien, K., *et al.*, A System of Film Badge Interpretation, *Nucleonics*, vol. 14, no. 10, p. 58, October, 1956). Calibration curves made with appropriate beta and gamma radiations then enable the two radiations to be evaluated separately. One point which should be carefully considered is that the beta plus gamma radiation through the "open window" often results in a density so high that density can no longer be considered proportional to exposure. In such a case, the exposure due to betas cannot be found merely from the difference between the filter and "open-window" densities (Bass, H., Beta Film Monitoring Procedure, *AEC Rept. NYO-1512*, 1950) and the actual calibration curve must be consulted.

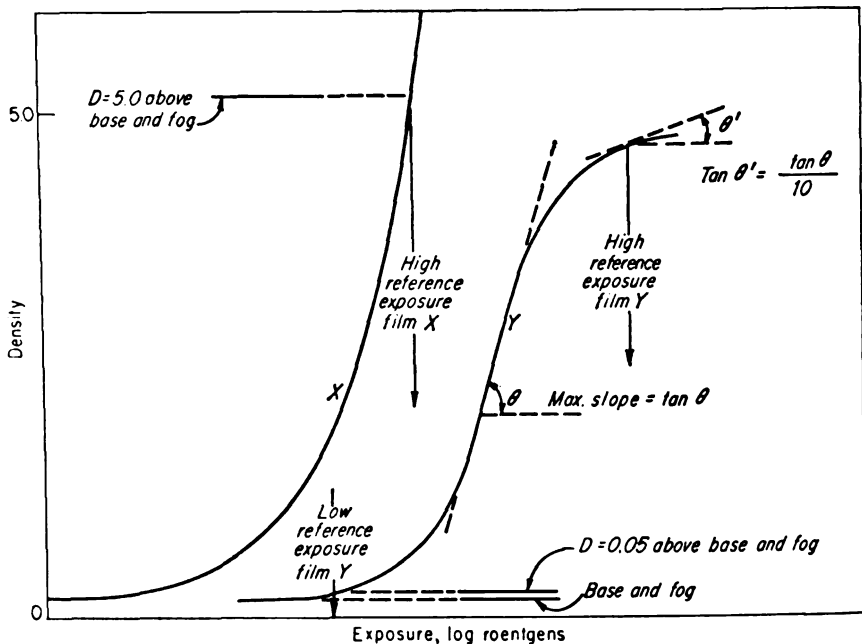


FIG. 10-41. Characteristic curves for two typical films used in radiation monitoring. The ASA criteria for *high-* and *low-reference exposures* are indicated.

The above method can, of course, be combined with the methods for simultaneous monitoring of X- and gamma rays of a wide energy range. If the badge used had, for instance, 1 mm cadmium, 1 mm aluminum, and "open-window" sections, hard X- or gamma rays would be recorded under the cadmium, softer photon radiation under the aluminum, and the beta radiation could be determined by the density in the "open-window" section. If, in addition, a cellulosic filter (about 30 mg/cm²) were added to the array, an estimate could be obtained of the effective beta-ray energy (Storm, E., *Response of Sensitive 552 Du Pont Film to Beta Radiation*, Los Alamos Sci. Lab. Rept. LA-1284, 1951).

Neutrons and Gamma Rays. Simultaneous thermal-neutron and hard-gamma-ray exposures can be measured by a similar procedure. If a film is exposed in a badge having cadmium and brass filters, the density under the cadmium will be a result of both the direct gamma-ray exposure and the neutron-induced gamma activity in the cadmium. Under the brass, the effect will be due solely to the direct gamma radiation. In addition, an "open window" will allow simultaneous estimation of any beta radiation. In large installations, where processing conditions are under strict control, and where the levels of exposure are such that the densities are on the straight portion of the *D* vs. *E* curve, a nomogram can be constructed to facilitate the evaluation of the densities from a gamma-ray calibration curve alone (Kalil, F., *Film-badge Measurement of Combined Thermal Neutrons and Gamma Rays*, *Nucleonics*, vol. 13, no. 11, p. 91, November, 1955).

PHOTOGRAPHIC FILMS FOR X-, GAMMA-,
BETA-RAY MONITORING

The **approximate characteristics of commercially available films** for use in radiation monitoring are given in Tables 10-4 and 10-5. Table 10-4 gives the speeds to 50-kvcp X-rays filtered through 0.5 mm copper plus 1.0 mm aluminum; Table 10-5 gives the speeds to radium gamma rays, filtered through 0.5 mm platinum (American Standard Methods for Evaluating Films for Monitoring X-rays and Gamma Rays Having Energies up to 2 Million Electron Volts, PH2.10/34, American Standards Association, New York, 1956). These tables indicate approximately the maximum and minimum sensitivities, respectively, of the films. The ASA criterion for **low-reference exposure** is that exposure in roentgens which will produce a developed density of 0.05 above base and fog density. The **high-reference exposure** is that exposure which gives a density of 5.0 above base plus fog (Fig. 10-41, film X), or that exposure at which the slope of the characteristic curve is 0.1 times the maximum slope (Fig. 10-41, film Y) (American Standard Methods for Evaluating Films for Monitoring X-rays and Gamma Rays Having Energies up to 2 Million Electron Volts, PH2.10/34, American Standards Association, New York, 1956).

The values in Tables 10-4 and 10-5 should be used only as rough guides in the selection of a film for a particular purpose and *should not be regarded as constituting calibration data*, since the response of a film is dependent upon exposure conditions and development.

CHEMICAL MONITORING OF RADIATION

Sanford C. Sigoloff

References: Hine and Brownell, "Radiation Dosimetry," Academic Press, 1956. Sigoloff, S. C., A Low Dose Range Chemical Radiation Detector for Personnel Monitoring, *Am. Ind. Hyg. Assoc. Quart.*, vol. 17, p. 426, 1956. Sigoloff, S. C., "Chemical Systems for the Measurement of Penetrating Radiation: Techniques and Production," USAF School of Aviation Medicine.

Chemical dosimeters utilize ionizing-radiation-induced chemical reactions to measure energy absorption. Yields of these radiation-induced reactions can be expressed in two ways: the energy yield G and the ionic yield M/N . G is the number of molecules of each product formed per 100 ev of energy absorbed in the reaction medium. M/N is equal to the ratio of the number of molecules changed to the number of ion pairs formed in the medium.

Radiation-induced reaction products are relatively stable and are easily identified in liquid systems when compared with gases or solids. Final products can usually be measured by precise analytical procedures.

DOSIMETRIC CONSIDERATIONS

A chemical reaction of known G value which is readily reproducible, relatively rate- and energy-independent, and with known temperature dependence can be used as a dosimetric system. Chemical systems that meet the above requirements usually have G values less than 20 and are, therefore, best suited for measuring kiloroentgen dose. Measurement of radiation doses in the 100- to 1,000-r range is accomplished by using systems with G values greater than 20. This necessary magnification is obtained using systems that proceed by a chain reaction. However, greater care must be exercised in achieving reproducibility.

Measurement of radiation doses between 500 mr and 3.0 r has been accomplished using systems with G values approaching 500. These systems are halogenated hydrocarbons such as chloroform or tetrachloroethylene overlaid with an acidimetric (pH-sensitive) dye.

EVALUATION TECHNIQUES

The halogenated hydrocarbons, chloroform and tetrachloroethylene, respond to radiation by liberating a spectrum of acids. The acids liberated can be utilized to reduce the pH, thus affecting an indicator dye. The acids can be measured by back

titrating with a standard base solution, by color comparison, by pH measurement, or by spectrophotometric analysis.

Spectrophotometric evaluation is the suggested technique for the following reasons:

1. This technique assures greatest accuracy.
2. There is less chance for human error.
3. Since the ampoules do not have to be opened as in titration, there is almost no chance for contamination.
4. Semipermanent records are available in the form of two graphs per dosimeter.

Spectrophotometric Measurements of pH Changes in Chlorophenol Red Indicator. Chlorophenol red has two peak absorption bands, 580 and 432 $m\mu$. As the pH of the dye is reduced from 6.0 to 5.0 (red to yellow), the percentage transmission at 580 $m\mu$ changes in an inverse relation. Since the mass-action law is applicable to the acid or alkaline forms of the dye, pH is a function of the ratio of the acid over the alkaline forms, and, conversely, pH controls the amounts of the two forms of the dye present in any solution of the dye. The precise evaluation of the pH from the ratio is accurate only if the dye concentration, tube size, light source, glass thickness, temperature effects, buffering action, and impurities are controlled and relatively constant.

Evaluation. Using a modified Beckman model DK-2 ratio recording spectrophotometer, it is possible to evaluate easily and accurately the above-mentioned ratio.

The DK-2 read-out modifications consist of special holder inserts to view only the center section of the dye system through a 0.070-in. hole. This hole size was chosen to give minimum glass-curvature aberration and an area large enough to reduce the effect of field spots and scratches. The ampoules are height- and area-positioned by hand. With allowances for changes in glass-wall thickness, inside diameter, and aberrations of the monprecision ampoules used, reproducibility can be as great as ± 3 per cent with reasonable economy. In addition, several changes in preparation techniques have reduced ampoule-production rejects from 15 to less than 5 per cent. Ampoules are centrifuged to 5,000 rpm prior to read-out to reduce glass-surface air bubbles and remove hydrocarbon droplets from the dye area.

Determination of Exposure Dosage. Dosimeters are produced at a chosen pH value and in the required sensitivities. They are then aged for possible leaching and are color-selected for uniformity.

Before exposure the dosimeters are evaluated for their preexposure ratio

$$\left(\frac{\text{per cent } T \text{ at } 580 \text{ } m\mu}{\text{per cent } T \text{ at } 432 \text{ } m\mu} \right).$$

See Figs. 10-42 and 10-43. Dosimeters are then given known dose exposures and are spectrophotometrically evaluated for their postexposure ratio

$$\left(\frac{\text{per cent } T \text{ at } 580 \text{ } m\mu}{\text{per cent } T \text{ at } 432 \text{ } m\mu} \right).$$

A curve of delta-ratio-units change (postexposure minus preexposure ratio) vs. dose is drawn. Once this curve is drawn, the delta-ratio-units change can be con-

verted directly to dose for one specific range of dosimeter. All the dosimeters must start at the bottom of the straight portion of the ratio vs. dose curve if linearity is to result. If accidental overexposure results, the dosimeters are not lost but can

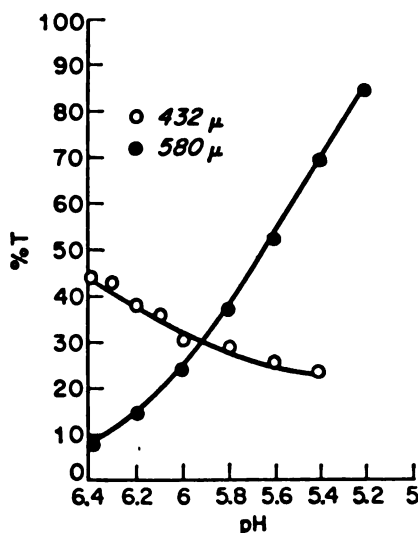


FIG. 10-42. Relationship of per cent T at 580 $m\mu$ and 432 $m\mu$ vs. pH.

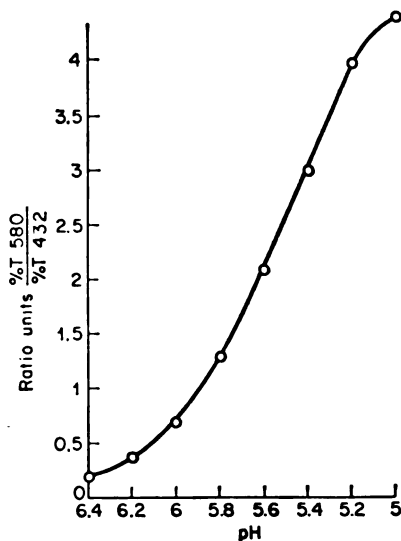


FIG. 10-43. Relationship of ratio $\frac{\%T_{580}}{\%T_{432}}$ vs. pH.

be evaluated by titration. Also, if a dosimeter is not exposed to its limit, it can be exposed again provided its total integrated dose range is not exceeded.

STATE OF THE ART

Chemical dosimeters of the personnel-monitoring type in the 500-mr to 3-r range have been produced and have been found acceptable. Shelf life, reproducibility, and sensitivity are reasonable. However, several laboratories are working toward improving these qualities.

Further investigation should produce a suitable dosimeter capable of visually indicating the approach of the tolerance dose or an immediate overexposure. In either case, the film badge will be read immediately for medical-legal records. If the overexposure is beyond the range of the film, the dosimeter is then titrated back for the "accident" dose. Under most conditions, a reasonably reliable dose can be obtained.

DOSIMETRY USING THE OPTICAL PROPERTIES OF SOLIDS

James H. Schulman

Exposure to ionizing radiation can produce a number of changes in the physical properties of insulating solids. Changes in optical properties—specifically in absorption and luminescence—are especially striking with some materials, and these effects have been extensively investigated for application to dosimetry. In general, absorption changes have been found most useful for the high-dose ranges (10^3 to 10^7 r), although it is possible to detect doses of the order of 10 r with specially sensitized materials in some circumstances. Luminescence effects vary considerably in their sensitivity ranges, some systems being capable of detecting the order of milliroentgens, while others can indicate in the range of 10^6 r or higher. Although absorption and luminescence phenomena are best investigated using large single crystals, the practical applications generally employ glasses or plaques of microcrystalline powders, because of lower cost, adaptability of these materials to various shapes, and ease of large-scale production.

All inorganic-solid dosimeters exhibit an energy-dependent response at low energies because of the comparatively high atomic numbers of their constituents. Selective shielding of these devices is normally necessary in order to achieve energy independence of the detector.

All the devices to be described are secondary instruments and must be calibrated against a primary standard to determine dose. Most of the studies have involved the X-ray and gamma-ray response of the devices, and little has been reported systematically on their response to electrons, alpha particles, or neutrons.

GENERAL THEORY OF OPERATION

When an electron is freed from an atom by irradiation of a solid, it may wander through the material until it is "trapped" by some other constituent, such as a chemical impurity, or by some region of the solid containing a flaw or defect in the structure. The entity, or "center," formed by combination of the electron with the trapping site has optical properties different from the normal crystal or glass; i.e., it may absorb light in spectral regions where the normal solid is transparent, it may emit fluorescent light under conditions where the normal solid would not do so, or it may fail to fluoresce under conditions where the normal solid would emit. The radiation-induced changes may be detected by optical techniques and are a measure of the dose.

If the electron is tightly bound to the impurity or defect, the center is a stable one; if the binding is weak, the electron may readily escape from the trap and return to its proper location in the solid. The center must be stable, at least at normal temperatures and in darkness, in order to be useful for dosimetry; otherwise the radiation-induced changes will be reversed on storage and the new optical properties will disappear. The centers need not be stable at high temperatures or under illumination, however; and indeed, certain methods of dosimetry make use of phenomena occurring when electrons are liberated from traps by heating or by illumination with infrared light. The latter methods, which expel electrons from their traps in the process of measuring the dose, are "one-shot" operations, the dose indication being necessarily erased by the process of measurement. Although the dosimeter may be reused to register new exposures, the original dose cannot be checked by repeated readings. When the measuring process does not destroy the centers, the dosimeter may be reread as many times as desired for confirmation of the original reading. The various types of optical effects used in dosimetry are summarized in Table 10-10.

Table 10-10. Optical Effects Employed in Solid Dosimeters

Type of centers induced by ionizing radiation	Measurement procedure	Effect	Description
I. Radiation-induced centers stable to measurement procedure	Illumination with light (generally ultraviolet or visible)	Coloration	New centers absorb light in spectral regions where solid was originally transparent
		Radiophotoluminescence	Unirradiated solid does not fluoresce under illumination with light used in measurement. New centers absorb measurement light and emit luminescent light at longer wavelength. Luminescence continues as long as measuring light is incident
		Degradation of luminescence	Unirradiated solid luminesces under illumination with measurement light. New centers do not luminesce under this light and diminish the normal luminescence. Luminescence continues as long as measuring light is incident
II. Radiation-induced centers necessarily destroyed by measurement procedure	Heating	Thermoluminescence	Energy stored in solid, in form of radiation-induced centers, released as luminescence. Luminescence decreases with time as sample is held at high temperature
	Illumination with light (generally infrared light)	Stimulated luminescence	Energy stored in solid, in form of radiation-induced centers, released as luminescence. Luminescence decreases with time as measurement proceeds

TYPE I DOSIMETERS

Coloration Dosimeters.* The operation of this type of instrument may be illustrated by reference to Fig. 10-44, which shows the absorption changes produced upon irradiation of a **silver-activated phosphate glass** (curves *A* and *B*). The absorption band due to the radiation-induced centers peaks at about 3,400 Å and extends into the region of the visible spectrum with sufficiently high exposure. The relationship between absorption and dose is shown in Fig. 10-45 for various wavelengths of measurement. The dosimeter element is a small polished plate of glass (about 5 to 10 mm on a side and 0.5 to 3 mm thick). The optical measurement is made with any suitable densitometer or spectrophotometer. The useful dose region for devices of this type is similar to that shown in Fig. 10-45. Alkali halide crystals specially sensitized with hydride † (KBr-KH or KCl-KH solid solutions) are considerably more sensitive than this, however, doses of the order of 10 r being detectable in crystals of reasonable size (1-in. length). Self-indicating step-type dosimeters for personnel-casualty monitoring have been proposed wherein the exposure is estimated by visual matching of the crystal coloration with colored reference strips simulating doses of 50, 100, 200, and 400 r.

Radiophotoluminescent Dosimeters.‡ Illumination with light in absorption band *B* of Fig. 10-44 excites the radiation-induced centers to emit luminescent light of the spectral distribution shown in curve *C*, Fig. 10-44. The intensity of the luminescence provoked by a given intensity of exciting light is determined by the number of centers in the solid, and hence is a measure of the dose. The silver-activated phosphate-glass dosimeter,¶ planned for personnel-casualty dosimetry for military personnel, covers the range of 5 to 600 r, the lower limit being set by the sensitivity and reproducibility of the glass and the impurities which give rise to a "predose" luminescence, while the upper limit is an arbitrary cutoff point de-

* Eicher, M., Naval Medical Research Institute (Bethesda, Md.), Report on Project NM 006 012.04.69, May 17, 1954.

Schulman, J. H., C. C. Klick, and H. Rabin, *Nucleonics*, vol. 13, no. 2, p. 30, 1955.

Rabin, H., and W. Price, *Nucleonics*, vol. 13, no. 3, p. 33, 1955.

Proctor, B. E., S. A. Goldblith, and S. Davison, *AEC Rept.* NYO-3345, 1955.

Davison, S., S. A. Goldblith, and B. E. Proctor, *Nucleonics*, vol. 14, no. 1, p. 34, 1956.

Birnbaum, M., J. H. Schulman, and L. Seren, *Rev. Sci. Instr.*, vol. 26, p. 457, 1955.

Fowler, J. J., and M. J. Day, *Nucleonics*, vol. 13, no. 12, p. 52, 1955.

Day, M. J., and G. Stein, *Nature*, vol. 168, p. 644, 1951.

Friedman, H., and C. P. Glover, *Nucleonics*, vol. 10, no. 6, p. 24, 1952.

† Friedman, H., and C. P. Glover, *Nucleonics*, vol. 10, no. 6, p. 24, 1952.

‡ Schulman, J. H., *NRL Rept.* 3736, September, 1950.

Schulman, J. H., Report of Symposium, IV, Chemistry and Physics of Radiation Dosimetry, Part II, Paper 5, p. 87, Army Chemical Center, Maryland, 1950.

Schulman, J. H., R. J. Ginther, C. C. Klick, R. S. Alger, and R. A. Levy, *J. Appl. Phys.*, vol. 22, p. 1479, 1951.

Schulman, J. H., W. Shurcliff, R. J. Ginther, and F. H. Attix, *Nucleonics*, vol. 11, p. 52, 1953.

¶ Schulman, J. H., W. Shurcliff, R. J. Ginther, and F. H. Attix, *Nucleonics*, vol. 11, p. 52, 1953.

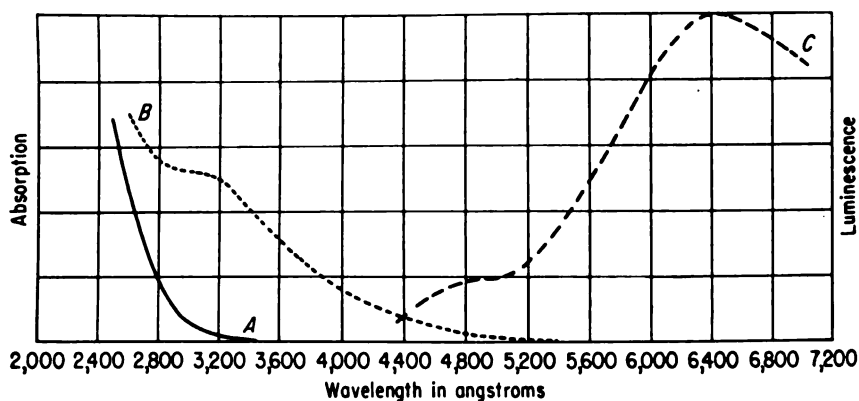


FIG. 10-44. (A) Absorption of undosed silver-activated phosphate glass. (B) Absorption of dosed silver-activated phosphate glass. (C) Luminescence emission of dosed silver-activated phosphate glass illuminated in spectral region encompassed by band B.

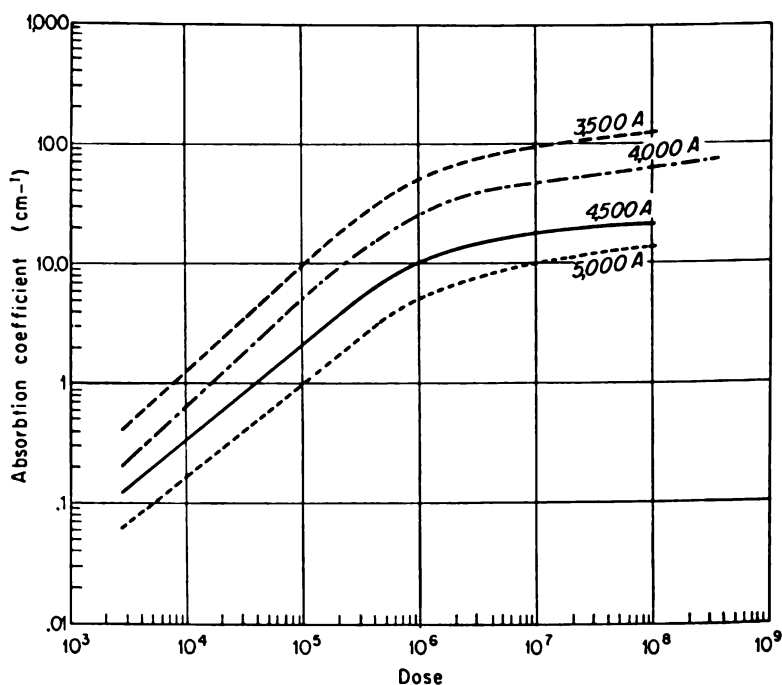


FIG. 10-45. Absorption coefficient vs. dose at various measurement wavelengths for silver-activated phosphate glass.

terminated by its function as a personnel dosimeter. The response of the device increases linearly with dose up to about 20,000 r, at which point saturation effects set in. The dosimeter consists of a rectangular piece of glass housed in a two-piece plastic locket, the whole assembly being about the size of a pocket watch and weighing less than an ounce. The dosimeter is rate-independent, cumulative, and retains its reading indefinitely after exposure despite repeated readings of the dose. The device is not "self-indicating" since means must be provided to excite the luminescence. Measurement of dose can be made either by visual observation of the luminescence or by photoelectric measurement. Line-operated and battery-powered fluorimeters have been developed, consisting of an ultraviolet lamp to excite the glass, a reference phosphor, and a photomultiplier tube with associated circuitry to measure the phototube current and indicate the dose.

An alternative form of the dosimeter * consists of a needle of the same glass 1 mm in diameter and 6 mm long. This miniature device is intended for insertion into tissue or body cavities for experimental radiobiology or therapeutic work. The lower limit of detection with the glass needle is of the order of 50 r. More intense exciting light is required in the fluorimeter to compensate for the smaller mass of glass used, but the operating principle and characteristics are otherwise identical to the personnel dosimeter.

Luminescence degradation occurs in certain solids, which normally luminesce under appropriate excitation. They have their luminescence ability destroyed by exposure to ionizing radiation.† This effect has been investigated for only a limited number of materials, mainly organic compounds such as anthracene and naphthalene.‡ The measurement technique is identical to that used for radiophotoluminescent dosimeters, but the decrease in fluorescence, rather than an increase, is the measure of dose. The dose range encompassed by this method is of the order of 10^5 to 2×10^8 r, with little indication that the effect will be found useful for dose ranges of interest in radiation hygiene.

TYPE II DOSIMETERS

In these devices electrons are liberated from their traps by heating or by illumination with infrared light, and luminescence occurs *on the return* of the electrons to the positions they normally occupy in the unirradiated solid.

For quantitative work, these dosimeters require an instrument to heat or illuminate the solid and to measure the integrated light output. These devices are inherently capable of very high sensitivity, the order of milliroentgens being detectable. The principal disturbing factor which precludes the practical use of the most sensitive devices of this type developed to date is the marked instability of the radiation-induced centers at normal temperatures. The number of trapped electrons decreases with elapsed time between the exposure and the reading, and the

* Schulman, J. H., and H. W. Etzel, *Science*, vol. 118, p. 184, 1953.

Etzel, H. W., R. D. Kirk, and J. H. Schulman, *RA-DET*, vol 8, p. 49, 1955.

† Schulman, J. H., *NRL Rept.* 3736, September, 1950.

‡ Birks, J. B., and F. A. Black, *Proc. Phys. Soc. London*, vol. A64, p. 511, 1951.

Black, F. A., *Phil. Mag.*, ser. 7, vol. 44, p. 263, 1953.

Schulman, J. H., H. W. Etzel, and J. Allard, *J. Appl. Phys.*, July, 1957.

dose indication is thus a function of this elapsed time. Further research may be expected to overcome this defect without loss in sensitivity. A dosimeter of this type would be a considerable improvement over the photographic-film badge in many respects and would play an important role in radiation hygiene.

Thermoluminescence dosimeters * operate on the principle that, as the temperature of the sample is raised, a luminescent glow appears, reaching a peak intensity at a temperature where the rate of electron liberation is a maximum. It then decreases to zero as the store of trapped electrons is depleted. If there are several different types of electron traps, the "glow curve" will have a corresponding number of peaks. The area under the "glow curve" is a measure of the number of trapped electrons and hence a measure of dose.

Thermoluminescent materials such as $\text{CaF}_2\text{:Mn}$ and Al_2O_3 have been developed which have long-term storage ability. Their sensitivities, however, are not high enough for day-to-day personnel monitoring, and dosimeters using them as detecting elements would be restricted to the casualty range (10 r and above). The more sensitive materials $\text{CaSO}_4\text{:Mn}$ and LiF suffer from loss of dose indication with storage time, but the order of 10 mr can be detected with the first of these two.

Stimulated luminescence dosimeters depend upon stimutable phosphors such as SrS:Eu , Sm ,† and NaCl:Ag ,‡ which have been used as the sensitive element in this type of dosimeter. The minimum detectable dose is of the order of milliroentgens, but the loss of dose indication on storage is a serious drawback. This "fading" is moreover a function of dose and dose rate for SrS:Eu , Sm .

GENERAL

The effects of X- and gamma radiation on the optical properties of solids have been applied to produce simple, rugged, and inexpensive dosimeters useful for casualty monitoring, for internal use in biological studies, and for high-range dosimeters useful in food and drug sterilization work. A number of promising devices have been developed for low-level personnel monitoring. Further research may be expected to provide practical devices for the latter range as well as dosimeters responsive to corpuscular radiations important in radiation hygiene.

* Daniels and W. P. Rieman, Final Report on Contract No. DA18-108-CML-3069, Chemical Procurement Agency, Feb. 16, 1954.

Ginther, R. J., and R. J. Kirk, "Thermoluminescence of $\text{CaF}_2\text{:Mn}$ " *Electrochem. Soc.*, vol. 104, p. 365, June, 1957.

Kossel, W., U. Mayer, and H. C. Wolf, *Naturwiss.*, vol. 41, no. 9, p. 209, 1954.

† Antonov-Romanovsky, V. V., *et al.*, Conference of Academy of Sciences, USSR, on Peaceful Uses of Atomic Energy, July, 1955, English translation prepared by U.S. AEC (1956), p. 239.

‡ Mandeville, C. E., and H. O. Albrecht, *Phys. Rev.*, vol. 91, p. 566, 1953.

Section 11

INDUSTRIAL APPLICATIONS *

By

EDGAR C. BARNES, B.S., *Manager, Industrial Hygiene Dept., Bettis Atomic Power Division, Westinghouse Electric Corp., Pittsburgh, Pa.*

Health-protection Practices.....	11-2
General.....	11-2
Air Sampling.....	11-4
Maximum Allowable Concentrations.....	11-7
Biological Sampling.....	11-9
External Radiation.....	11-10
Periodic Physical Examinations.....	11-12
Air Pollution.....	11-12
Water and Stream Pollution.....	11-14
Natural Radioactive Materials.....	11-16
Reactor-core Manufacturing.....	11-20
Reactor Operation and Materials.....	11-25
Industrial Radiography.....	11-35
Gamma-ray Sources.....	11-37
Sealed Beta-ray Sources.....	11-42
Radioactive Isotopes.....	11-45
Thorium.....	11-50

* Assistance in suggesting and preparing text was given by: Duncan A. Holaday, U.S.P.H.S., Salt Lake City, Utah (*Natural Radioactive Materials*); Paul R. Bolton, Bettis Atomic Power Division, Westinghouse Electric Corp., Pittsburgh, Pa. (*Reactor-core Manufacturing*); H. Floyd Herr, Bettis Atomic Power Division, Westinghouse Electric Corp., Pittsburgh, Pa. (*Gamma-ray Sources and Sealed Beta-ray Sources*); W. David Small, Bettis Atomic Power Division, Westinghouse Electric Corp., Pittsburgh, Pa. (*Reactor Operation and Materials*).

INDUSTRIAL APPLICATIONS

Edgar C. Barnes

HEALTH-PROTECTION PRACTICES

General

See page 11-52 for General References.

Toxic or poisonous materials, of a wide variety, are used in many different industries. Any of these materials can be used in such a manner that no injury or disability will occur to the employees. Situations can arise, however, which will produce illness, disability, or death.

When employees are rapidly and acutely disabled by exposure to such materials, the situation is usually referred to or defined as an **accident**. **Workmen's compensation laws** are in effect in all 48 states. These laws require the employer to pay medical expenses and a stipulated rate to employees while disabled by an accident. They provide for payment of specific amounts to beneficiaries in case of death.

When employees develop a chronic disease from exposure to such materials, or where repeated exposures produce a disease, the situation is usually referred to or defined as an **occupational disease**. Some states have enacted occupational-disease acts which provide for payment to employees in a variety of ways for these disabilities. If not covered by such laws, employees may enter claims against the employer in court for losses due to disability or loss of earning power.

Many states have established specific **state standards or codes** which require industries in that state to meet certain minimum requirements for safety or health protection. Sometimes these are laws passed by the state legislative bodies, but usually they take the form of regulations, standards, or codes promulgated by health or labor departments under general authority granted by law. The cognizant department can enforce them by inspection and issuance of orders requiring compliance. A further review of regulations by states can be found in Sec. 3.

The financial liability and the requirement of meeting specific safety or health regulations, along with a basic philosophy that health of employees should be maintained at the best possible level, provide a very strong incentive for industry fully and completely to eliminate any situations which can cause occupational diseases or accidents.

The basic emphasis in industry must always be on prevention of injuries, rather than on medical treatment or other "after-the-fact" control measures. However, this basic approach does not minimize the need for adequate medical personnel and

facilities to treat persons who may have received an injury or contracted an occupational disease because of inadequate protection. A complete **industrial medical department** is essential both for treatment of injuries and for conducting preplacement and periodic physical examinations (see Sec. 23). Such examinations ensure that there will be no unsuspected effects on health and guarantee that the plant control measures are providing the desired health protection.

The **prevention of accidents or occupational disease** is accomplished by carefully avoiding exposures of sufficient magnitude to produce any detectable physiological reactions in employees. Toxic or poisonous materials may gain entrance to the body by breathing, ingestion, or absorption through the skin. Radioactive materials can be considered as being in the same category as toxic or poisonous materials in so far as they may produce unfavorable physiological reactions if they gain access into the body. When radioactive materials gain access into the body, the resulting irradiation of tissue from these sources is called **internal radiation exposure**. The physiological reactions come as a result of the radioactive property of these materials, whereas nonradioactive materials produce their toxic reactions by chemical means.

When considering health hazards from radioactive materials, however, the added factor of **external radiation exposure** must be included. Radioactive materials or other radiation sources which remain outside the body at all times can cause irradiation of the body tissues, which may result in unfavorable physiological reactions when the exposure is sufficient. Thus, when health-protection measures involving radioactive materials are established, the control measures must take into account the factors of internal exposure such as inhalation, ingestion, and skin absorption, as well as the control of body irradiation from sources external to the body. In the case of radiation sources other than radioactive materials, such as X-rays, only the external radiation exposure factor exists.

In industry, the major **route of entry** into the body for toxic or radioactive materials is the respiratory system. Factors of ingestion and skin absorption cannot be neglected because some specific materials may gain access to the body very readily by these means. The major problem, however, is to control the inhalation of dusts, fumes, gases, vapors, and mists.

To prevent **inhalation** of toxic or radioactive materials, the first approach is usually to design the process and equipment in such a way that none of the material escapes into the air which is to be breathed by workmen. This is accomplished by such means as elimination of dust-producing operations, proper selection of the chemical and physical form of the material to be processed, and total enclosure of processes. Where such fundamental controls cannot be applied, it becomes necessary to use some means of control such as local exhaust ventilation. Finally, where these methods are inadequate, the employees must be provided with some means of personal protection such as respirators. For further information, see Secs. 17, 18, and 22.

In addition to controlling the hazards among employees in industrial plants, industry is frequently required by law, or by a sense of moral values, to control the emission of any effluent from its plant which would create a nuisance or hazard in areas or communities adjacent to its plant location (see Sec. 3). Air-pollution control is so important that it has been the subject of much legislation, and industry

must ensure that its operations do not contribute significantly to air contamination. Any hazardous constituent in liquid effluent must also be kept within permissible limits, and hazardous or nuisance amounts of toxic or radioactive materials should not be released in either scrap or articles of commerce.

Air Sampling

Where toxic or radioactive materials are used in any form which can produce dust, fumes, gases, vapors, or mists, it is usually desirable to determine their concentration in the air by means of **air sampling**. When the concentration of the air contaminant has been determined, this value is then compared with a known **hygienic standard** or **maximum allowable concentration** to determine whether breathing of this atmosphere, under the conditions that prevail, may result in sufficient exposure of employees to produce a health hazard.

A wide variety of **air-sampling devices** is available commercially, and many "homemade gadgets" have been made. The nature and concentration of the air contaminant usually dictate the best method of sampling, but considerable judgment must be exercised so that the collection of the sample and the subsequent analysis actually provide a reasonably accurate evaluation of the concentration of the particular substance in the air. Instantaneous or short-time sampling may lead to completely false interpretations of the true average exposure of personnel. Also, samples taken at any location other than the breathing zone of a workman should not be assumed to represent his exposure. Furthermore, samples taken on any one day can be far from representative of conditions on another day. Ingenuity is required to evaluate exposures by air sampling, except when long-time samples are collected on many different days and they represent the air actually breathed by workmen. Correct interpretation of air samples is extremely important.

Filtration devices are commonly used for air sampling. A source of suction such as a pump, an air-flow-measuring device, and a holder containing a filter paper or membrane may be used. Complete units of this type are commercially available, and a variety of individual components can be obtained and assembled for this purpose. Both filter papers and plastic membranes are used. The latter have a higher collection efficiency and the dust is retained on the face of the membrane. In the case of filter papers, the radioactive particles become embedded in the papers, and absorption of alpha particles in the paper may give an inaccurate count.

Electrostatic precipitators provide a highly efficient method of air sampling for dusts, fumes, and mists. Units are commercially available which consist of an electrostatic sampler head that deposits the particulate matter on the inside of an aluminum tube, and a source of suction which is calibrated to give a specific air flow. Many homemade electrostatic precipitators have been designed for specific applications and are mentioned in the literature.

Adsorption devices for air sampling are used for sampling of a variety of vapors and gases. A typical assembly consists of a tube filled with activated charcoal, a flowmeter, and a blower to provide air movement through the sampling tube.

Impingement sampling devices are used for such air contaminants as dusts and fumes. These may consist either of devices such as a Konimeter, Owens Jet dust counter, or annular kinetic impactor to impinge the particulate matter directly

onto a surface in such a manner that the solids are retained on the surface, or of devices which combine impingement and a wetting process, such as the Greenberg-Smith apparatus or the midget impinger. The **annular kinetic impactor** has been found useful for sampling radioactive particles in the presence of a radon or thoron background without the necessity of waiting for the radon, thoron, and their daughters to decay (Tait, G. W. C., Determining Concentration of Airborne Plutonium Dust, *Nucleonics*, vol. 14, pp. 53-55, Jan. 1, 1956).

Thermal precipitation devices are used for sampling dust, fumes, and mists. In these devices the particulate matter is deposited on a surface by passing the air between a heated and a cold surface. These devices usually operate at a very small rate of air flow.

Sedimentation devices have been used in a limited manner for air sampling. Such devices usually consist of a chamber in which a sample of air can be trapped. After it is trapped, the particulate matter settles onto a prepared surface which can subsequently be removed for analysis. One form of sedimentation device used extensively for sampling radioactive materials is the **fall-out tray** consisting of a 1-ft-square plate covered with a thin sheet of adhesive plastic (Eisenbud and Harley, *Science*, vol. 24, no. 3215, Aug. 10, 1956.) The fall-out trays are placed outdoors and the plastic is removed either daily or weekly. After ashing, the collected radioactivity is usually quantitated by counting procedures. The number of particles can, however, be determined by laying the exposed plastic sheet against a sheet of X-ray film and allowing an exposure of 1 day to 1 week. After developing, the number of darkened spots on the film is determined. This method also permits an estimation of the radioactivity in individual particles by an evaluation of the size and blackening of the spots produced on the film. These methods have been particularly useful for determining the fall-out of radioactive materials from weapons tests. By careful evaluation and interpretation, the fall-out tray can be used for detecting and determining the radioactive materials that precipitate around a plant which is processing radioactive materials.

Condensation devices may be used for collecting vapors or gases from the air. The air is drawn through a suitable chamber immersed in a low-temperature bath. The temperature must be considerably below the boiling point of the gas or vapor, and preferably below its melting point. One special application of this principle is that of sampling from a stack or other discharge, where the air or gas is at an elevated temperature and has a high moisture content. Withdrawal of a sample through a simple water-cooled condenser will frequently result in a highly efficient removal of both particulate and vapor constituents.

Vacuum bottles (or containers which permit flushing air through them and trapping a sample by means of valves) may be used to collect a sample of air which can subsequently be analyzed by a variety of techniques. They are usually useful for collection of gases (particularly inert gases such as radon) and, to a limited extent, vapors.

Filtration, electrostatic precipitation, and adsorption devices are most commonly used for sampling radioactive materials, but there are a variety of others, some of which involve direct determination by means of some physical, chemical, or radioactive property of the material. Such devices include the interferometer, portable Orsat, combustion devices, ultraviolet-absorption devices, and chemical indicators

either in liquid media or on papers. These, however, have limited application in determining the concentration of radioactive materials in the air unless some chemical or physical property of the material can be utilized.

Analysis of air samples of radioactive materials may be done by a wide variety of techniques. Frequently the radioactive property of the air contaminant may be used as a means of evaluation such as inserting the sample in counting equipment which senses each radioactive disintegration and counts the number of these. In some cases it is practical to make use of the physical or chemical properties of the radioactive material, and the analysis may be done by such methods as chemical analysis, spectrometry, dust counting, or use of the dropping-mercury electrode (see Sec. 17).

Counting equipment is commonly used for determination of the amount of radioactive material which has been obtained by any of the sampling methods outlined above. Samples collected on filter papers or membranes are usually analyzed by inserting the filter paper or membrane in (1) a lead pig containing a G-M tube, (2) a gas-flow proportional counter, or (3) a scintillation counter. Electric pulses from these devices are fed through scalers which count the number of pulses in a given length of time. These devices must be suitable for indicating the specific type of radiation given off by the material under investigation. More details of these devices are given in Sec. 17.

In general, it can be said that the same methods of analysis are used for samples of radioactive material obtained from the air as for other radiometric analyses. Frequently, however, extremely small amounts of material are involved; this necessitates great sensitivity, with a reasonable accuracy in the counting equipment. High accuracy is not required in the analysis of air samples for control of health hazards. Results within ± 10 per cent are usually adequate because the variability of the conditions where the sample is taken prohibits the collection of a sample that is truly representative of personnel exposure. Hygienic standards are seldom more accurate than this.

A continuous indication of the concentration of an aerosol can be obtained with a number of different devices. They usually consist of a **continuous-sampling device** coupled with a continuous analytical device and are arranged so that the readings indicating the concentrations are presented on a dial or chart. Such devices may incorporate an arrangement to warn of high concentrations in the air. Although a number of different devices have been tried for radioactive aerosols, the most common one consists of a moving filter paper arranged so that air is drawn through the paper as it moves over a port. The paper then passes under a sensing element, such as a G-M tube or other radiation-sensitive device. The number of counts per minute indicated by this sensing device is shown on a microammeter, and by proper calibration, this device can be used to indicate the concentration of radioactive materials in the air. Such devices are commercially available from a number of different companies and can be obtained with a variety of different characteristics to suit different applications. Such equipment is usually large enough that it is necessary to mount it on wheels or some other arrangement to make it movable. It is usually used in such a manner as to indicate the concentration of the radioactive aerosol in some preselected location in a workplace, and it does not indicate the average exposure of personnel. Some care must therefore be used in interpreting

data from the continuous air sampler. Many of these devices operate in such a manner that the data presented on the meter are indicative of the concentration in the air from 10 to 40 min previously. This delay in presenting the data must always be given consideration if the device is used in connection with an operation which might produce a very high concentration of the aerosol in a very short period of time. Significant exposures might occur before the device would give its indication.

Maximum Allowable Concentrations

It is a fundamental concept that the physiological reaction to a specific material is proportional to the amount of that material which is present in the physiological system involved. The higher the concentration of the particular material in the tissues or fluids, the faster and more severe the reaction. Theoretically there is a concentration of each particular material below which the physiological reaction is insignificant, and, for all practical purposes, it can be said that no unfavorable physiological reactions occur. In turn, the amount of any specific material that gains entrance to the body can be limited so that its concentration in the body tissues is low enough to avoid harmful reactions.

Since the major **route of entry** for toxic or radioactive materials used in industry is the respiratory system, it is possible to control the extent of the toxic reaction by limiting the amount of the material that is present in the air. One could adopt the attitude that the concentration of any potentially hazardous material in the air should be kept at zero and would then be assured there would be no reaction from it. However, this is impractical and unnecessary. As a workable solution to the problem, "maximum allowable concentrations," or hygienic standards to control occupational disease and accidents, have been established. The maximum allowable concentration for a specific material is defined as that concentration in air which can be breathed by workmen during 40 hr of work per week for an indefinite period of time without producing any detectable or significant pathological effect. This definition must be considered as the ideal because there are many substances which have not been studied in sufficient detail to ensure that the specific maximum allowable concentration will fully and adequately meet this criterion. Where adequate information is not available, the persons responsible for publishing values of the maximum allowable concentration have consistently applied safety factors which they feel are adequate to ensure that the requirements of this definition are met. Appropriate use of **periodic physical examinations** is a very real adjunct to control of hazards by use of maximum allowable concentrations. They can be used to confirm that the existing concentrations do not produce any detectable change in the physiological systems of personnel. Various other terms have been used to signify the same concept as "maximum allowable concentration." Such terms as maximum permissible concentration and safe limit have about the same connotation. It has recently been suggested that "hygienic guides" would better express the intent (Smyth, *Am. Ind. Hyg. Assoc. Quart.*, vol. 17, pp. 2, 129, June, 1956).

In the case of radioactive materials, it seems to be necessary to establish a maximum allowable concentration for each **chemical compound** of each particular isotope because the chemical behavior of each different compound determines its uptake in

the body. Different compounds of the same element may be taken up in quite different ways, depending upon such factors as valence, solubility, and particle size. Experience has indicated, however, that each element has its own characteristic behavior in the body, with the characteristic behavior being modified by such factors as solubility and valence. Frequently, the valence of the element when it enters the body is an important factor. The maximum allowable concentrations for the same element in different valence states may be quite different.

There are definite limitations in the present data on maximum allowable concentrations. Because it has not been practical to develop a maximum allowable concentration for each chemical compound of each isotope, it is customary, at present, to specify the maximum allowable concentration for each isotope, with only occasional reference to a particular chemical compound involved. It then becomes necessary for users of these data to interpret whether the particular chemical compound involved in their exposures will fall within the limits established for the element. In the case of radioactive materials, the tremendous amount of toxicological research which is required to establish a reasonably accurate maximum allowable concentration will require a long period of time. As an interim measure it has therefore become customary to calculate the maximum allowable concentrations based on available data regarding the distribution of the particular element in the body after it has been taken into the body by mouth or by inhalation. Formulas for such calculations will be found in Sec. 14.

It must be emphasized that these are only crude approximations and must be used with considerable judgment until such time as the biological reactions to these materials are more fully known. Periodic physical examinations must be made to detect any possible reactions where any significant exposures to radioactive materials occur. The best available information on the maximum allowable concentration for radioactive materials is published in NBS Handbook 52, which is periodically revised to reflect the latest information. For details, see Exposure Standards in Sec. 3.

Many **inert radioactive gases** such as Argon-40 emit only beta or gamma radiation. When these are present in the air, the external radiation exposure will usually be the controlling factor rather than internal radiation exposure. If the volume of gas surrounding a person is small, as in a small room, the maximum allowable concentration would be greater than that for exposure in an infinite volume. A formula for calculating this exposure, where the body can be considered as being immersed in an infinite volume of the radioactive fluid, is given in Sec. 3.

To summarize briefly, the concept of controlling exposure by use of maximum allowable concentration is very useful. When adequate data are available, the control of occupational disease or accidents by limiting the concentration of the particular material in the air to this value will result in good control. The number of radioactive materials for which adequate data are available is limited, and the maximum allowable concentrations should therefore be used judiciously with full recognition of their limitations. Other means of evaluation such as biological sampling and periodic physical examinations should, therefore, supplement this means of control.

Biological Sampling

Although numerous steps may be taken to ensure that employees will not receive overexposure to toxic or radioactive materials, it is frequently desirable to evaluate the actual exposures that have occurred. One of the most satisfactory means of measuring the true internal exposure of each employee is by evaluation of the amount of the specific material that is present in his body, or the amount of the material that is passing through his body. **Biological sampling** therefore becomes an extremely valuable adjunct to all other means of exposure evaluation which may have been established.

In industry, the most common means of biological sampling is to collect **urine samples** and analyze them to determine the concentration of one or more specific toxic elements. As a general rule, it can be said that toxic or radioactive materials which enter the body of employees by inhalation, ingestion, or absorption through the skin are metabolized in various ways by the body, but these materials or degradation products from them are always excreted in the urine. The metabolism of various radioactive materials in the body depends upon their chemical behavior rather than their radioactive properties. However, the specific isotope or its daughters taken into the body will be found in urine samples. Different chemical compounds of the same element may be metabolized in different ways, depending upon solubility and other physical or chemical properties. Different elements will likewise metabolize in different ways. Although this subject is entirely too complicated to elaborate on it here, in many instances an evaluation of the rate of excretion of a particular isotope or element in urine can be considered a satisfactory index of the exposure that persons may be getting in their industrial environment. Although the analysis of urine samples is by far the most common means of evaluating exposures, collection and analysis of feces, blood, or breath can be useful. Under special conditions, samples of hair, fingernails, or biopsy specimens may be used.

Urine-sample collection must be undertaken with considerable care. The samples must be collected under conditions which will ensure that there will be absolutely no contamination of the sample with the material for which the analysis is to be made. It is highly desirable to collect all samples under the direct supervision of a physician or nurse who must make certain that hands, clothing, or equipment are clean. It is the best practice to have such samples collected in a medical department or hospital. It is frequently desirable to obtain as large a urine sample as possible so that greater accuracy can be obtained in the analysis. It is also desirable to have the sample represent excretion over as long a period of time as possible, so as to minimize variations which occur in the excretion rate at different times of the day or because of metabolic variations. Urine samples representing a total excretion of periods of 24 hr or longer may be highly desirable; but usually it is impractical to obtain them from industrial workmen with any assurance that they are uncontaminated and that they are truly a complete sample, unless the person is hospitalized or otherwise controlled. In industry it is usually the best practice to collect the urine samples in the medical department during working hours, making certain that the sample represents total excretion over a period of several hours. It has been found practical, in many instances, to collect the urine samples repre-

senting total excretion of a measured period of time by having the employee report to the medical department where a urine sample is collected and the time is recorded. He then returns once or twice until a reasonable volume of urine has been collected and a reasonable period of time has elapsed. The time of the last voiding is noted. From these data the excretion rate per hour of some particular element can be evaluated, or it may be calculated as excretion rate per 24 hr. This method of sample collection also permits calculation of the concentration of the element in the urine for comparison with the rate of its excretion.

Exposure to the following radioactive materials has been evaluated by measurements of urinary excretion: uranium, plutonium, radium, thorium, strontium, ruthenium.

Biological sampling can be performed for two distinctly separate purposes: (1) to determine the **rate of intake** by personnel, and (2) to determine the **body burden** of some individual after an exposure. When people are working around radioactive materials and are breathing dusts from these materials, it can be considered that a portion of this material which is taken into the body is metabolized and excreted rather rapidly. Another portion of this material finds its way into storage sites where it may become immobilized or its rate of turnover becomes very long. That portion of the material which is excreted rather rapidly can be used as an index of the amount of material that a person has taken into his body. Collection and analysis of urine samples from persons during periods of active work will give an approximate indication of the amount of materials that are getting into different persons' bodies. When a person stops his exposure completely, the excretion of the material in urine becomes a reflection of the material that has been stored in the body. When exposure has ceased, the amount excreted in the urine each day will decrease over a long period of time. The rate of this decrease and the amount of material actually excreted can both be used in evaluating the "body burden" of the person. The amount of data available which can be used to aid in evaluating the body burden of the person for radioactive materials is very limited, but the work of Langham (*Am. Ind. Hyg. Assoc. Quart.*, vol. 17, no. 3, p. 305, September, 1956) gives an indication of the usefulness of this technique. It is frequently possible to make estimates of the body burden for any particular radioactive element with sufficient accuracy to indicate whether it is extremely low, and perhaps insignificant, or high enough that more careful evaluations must be made along with close medical follow-up on the person.

External Radiation

External radiation is defined as that radiation which penetrates body tissues from sources outside the body. It is the one additional factor, when dealing with radioactive materials, that must be integrated into an over-all health-protection plan. It is the only factor that is operative when dealing with X-rays or sealed sources of radioactive materials such as radium or Cobalt-60. Alpha radiation is not significant external to the body, but all others (beta, gamma, neutron, and proton) will penetrate and interact with body tissues. The interaction of either X-rays or gamma rays with tissue can be considered to be the same. Section 19 provides the factual data regarding the interaction of these radiations with tissue and matter.

It is a basic concept that effective health protection will be realized if body tissues, other than the skin, hands, forearms, feet, and ankles, do not receive an average radiation dose greater than 5 rems/year throughout a working lifetime (age eighteen to seventy). Higher exposures are permitted for the skin and body extremities. Also, higher exposures of tissues may be permitted for short periods of time. Section 3 should be consulted for complete rules. This limitation of tissue irradiation should include that part contributed by both internal radiation and external radiation. The determination or measurement of the total radiation dose to any particular tissue from both internal and external sources can become an extremely complicated matter. A practical and conservative method of controlling radiation exposure where both internal and external exposures exist is to add two exposure factors, where each factor represents the percentage of the maximum allowable exposure level.

$$F_1 = \frac{\text{average breathing exposure (microcuries/cc air)} \times 100}{\text{maximum allowable concentration (microcuries/cc air)}}$$

$$F_2 = \frac{\text{average external exposure (rem/year)} \times 100}{\text{maximum allowable exposure (rem/year)}}$$

$$F_1 + F_2 = \text{per cent of permissible radiation exposure}$$

Where **biological-sampling** methods have been used and good estimates of exposure to radioactive materials (or of specific tissue dose) have been obtained by this means, the factor F_1 can be obtained from this source. For example, urinary excretion of a radioisotope may indicate an exposure rate of one-quarter of the permissible level. F_1 would then be 25. The external radiation exposure can be obtained from film badges and/or dosimeters, from which F_2 can be calculated.

It is frequently practical to assign specific factors for certain types of work. For example, work with uranium highly enriched in the 235 isotope can be assigned $F_1 = 100$ and $F_2 = 0$, since external radiation exposure in this type of work is negligible. For work with natural uranium (which has been separated from its daughters) F_1 might be assigned a value of 90 and $F_2 = 10$. "Hot-lab" work might be set up on the premise that $F_1 = 20$ and $F_2 = 80$. For power-reactor operation F_1 could be 10 and $F_2 = 90$. Application of these limits should be based on average dosage over a period of time—not doses of a few hours or days.

In many instances, the additive procedure described above indicates a higher tissue dose than would actually be occurring. Also, the over-all biological effect may not be additive. Although a synergistic effect might be postulated, there is little evidence that it is significant. For these reasons, there is a growing tendency to apply the limits for internal and external radiation exposure independently and to keep each type of exposure well within the limits specified.

Neutron exposures should be added directly to other external exposures, when both are in rem units. Methods of measuring neutron exposure are given in the sections on dosimetry (Sec. 10).

Periodic Physical Examinations (See also Sec. 23).

Where persons work with radioactive materials, or where they may be exposed to external radiation, **periodic physical examinations** are usually desirable. About the only exception is in situations where experience over a period of years has clearly indicated that the exposure to the radioactive materials or radiation is small enough so it can be clearly stated that no significant exposure occurs. **Preplacement physical examinations** evaluate a person's physical condition in relation to the work that he will perform, indicate any unsuspected illness, determine the result of any previous exposure, and form the basis for subsequent evaluation of exposure during periodic physical examinations after the person has started to work on a particular job. With a satisfactory preplacement examination, a good basis has been laid for detecting any physiological changes that may come about because of industrial exposures, and also for detecting any "normal" physiological changes which may be occurring as a result of factors unrelated to employment. The type of physical examination to be given at periodic intervals will be determined by the physician on the basis of the physiological changes which may occur from exposures to the specific materials used in connection with the person's work. The frequency of such examinations is usually determined by an estimation of how great an exposure the person may get on his job. Where it is known that exposures will be limited to small percentages of the permissible levels, the frequency of the periodic physical examination may be 2 or 3 years. Where potential exposures are high in relationship to hygienic standards, or where new materials of unknown toxicity are involved, the examinations may be given monthly or, in rare instances, weekly. Probably the most commonly used frequency for periodic examinations in industry is yearly, except where conditions definitely indicate a potentially high exposure. The physician who is to make the periodic physical examination must be fully cognizant of the toxic or radioactive materials involved in the person's exposure, the nature of any external radiation exposure, and the nature of the work which the person is performing. It is usually desirable and practical to collect urine samples or other biological samples at the time of the periodic physical examination, although such biological sampling may be done more frequently. Close cooperation between the physician and the industrial hygiene engineer or health physicist must be realized to obtain the most satisfactory result from periodic physical examinations.

Air Pollution

Air pollution is a broad term which implies the vitiation of the atmosphere with any type of contaminant either in a restricted location or over a broad area. Although this term, as related to industry, has been used to include the many problems inside buildings, factories, or mines, the more common usage relates to air outside such places, and it is so used here. The atmosphere normally contains some dusts, mists, vapors, and gases which are generated and maintained there by natural causes. Dusts from wind erosion of the soil and particles from "salt spray" along the ocean are common examples. In the case of radioactivity, there are very significant amounts of radioactive substances normally present in the air. This **normal**

background may interfere significantly with measurements of the radioactivity added to the atmosphere by industrial processes. The inert gases radon and thoron, with their radioactive daughters, account for most of this normal radioactivity in the atmosphere. These gases are continually fed into the atmosphere from decay of uranium and thorium in soil and minerals. Concentrations may vary from only a few disintegrations/min/m³ to as much as 1,000 disintegrations/min/m³. Trace amounts of dusts from naturally occurring radioisotopes such as uranium, radium, thorium, and potassium will be present.

Air pollution from industrial processes may present problems involving effects on humans, animals, vegetation, buildings, and other structures, sensitive industrial processes, and sensitive items of commerce. Effects are extremely varied depending upon the contaminant involved, the concentration, and the exposure time. Air pollution from radioactive materials will, however, exert its effect mainly upon any living systems in the animal or vegetable world, on sensitive industrial processes, and to a limited extent, on sensitive items such as photographic film or scientific instruments. Air pollutants not only exert their effect while suspended in air, but many will settle out slowly, and effects from this settled material must be considered.

The full impact of adding radioactivity to living systems is not adequately understood and probably will not be fully understood for many years. A sufficient amount of scientific knowledge is available, however, to permit a rational analysis of this problem and to establish limitations which are adequate and probably conservative.

The National Bureau of Standards (for the National Committee on Radiation Protection) has listed in NBS Handbook 52 (see Sec. 3) the maximum permissible concentration of various radioactive isotopes in air for continuous exposure 24 hr/day, 7 days/week for an indefinite period. These concentrations are established on the basis that they will not produce any detectable biological damage. The basis for establishing these values, however, is their effect on normal adults such as those persons employed in industry. Since air pollution involves exposure of the general population, including all age groups and persons having a variety of diseases, the maximum permissible concentration for control of air pollution must be set at much lower values for regulating the release of radioactivity to the general environment. Furthermore, where the ultimate disposition of the air contaminant is not adequately defined with the relation to population groups, the limits are applied at the point where the contaminated air leaves a controlled area. These limits are usually set at one-tenth of the values given in Table III, NBS Handbook 52, and these lower values can be used as the "maximum permissible concentration" in the air which will be breathed by large population groups. They do not reflect possible effects on animals or vegetation, sensitive industrial processes, or sensitive articles of commerce. It is generally accepted that these other effects will be of a minor nature, where the air concentrations are kept below the limits indicated.

Primarily, industrial users of radioactive materials must ensure that any **release** of these materials will not produce a concentration in the air outside their buildings or outside their control which is above the established safe limits, when averaged over a period of a year or longer. Instantaneous release of very high concentrations should be avoided, even though they might be within the limits when averaged

over a year. If air containing radioactive materials in excessive concentrations is to be exhausted from industrial processes or buildings, some means of treatment, filtration, or dilution will be required. A wide variety of air-cleaning equipment is commercially available. See Sec. 22 for types and engineering information.

Some states, counties, and cities have established **codes** or **regulations** to control air pollution (see Sec. 3). Every industrial plant should determine what is legally required in its location. These controls may be administered by a department of labor, department of health, or a separate air-pollution-control agency. In addition to being governed by specific controls of such agencies, industrial plants may be held liable through court action for damages caused by pollution of the air or contamination of the environment by their operations. Since relatively large sums of money may be involved, either in the cost of control measures or in damage claims, a careful engineering and fiscal evaluation of air-pollution-control measures is essential in industrial operations.

Water and Stream Pollution

Stream pollution is a broad term which implies the vitiation of water in rivers, creeks, and lakes. All such waters normally contain particulate matter and dissolved compounds. They also support an extensive and complicated aquatic life ranging from the simplest organisms to fish, etc. Sometimes, naturally occurring minerals will contaminate a stream or body of water to the extent that it will not support aquatic life or supply water fit for human consumption. Mining and industrial wastes have also been permitted to pollute streams, but effective control of stream pollution is becoming a legal requirement in many locations.

Practically all streams contain radioactive materials. The amounts of these and the particular isotopes may vary over a wide range from time to time. Radioactive elements usually occurring naturally are uranium, thorium, radium, potassium, and decay products of these (see Secs. 4 and 6).

Industries which use, process, or produce radioactive materials may introduce radioactive materials into streams through their **plant liquid effluent**. This effluent may reach small streams and sewers by direct discharge; or it can be discharged into the ground where it may be transported by ground waters.

Where only a small amount of radioactive material is involved in the plant (such as specified in AEC Regulations published in the *Federal Register*; see Sec. 3) or where only sealed sources are used, the amount of radioactive material in the plant liquid effluent will usually be insignificant. However, continuous or repeated release of radioactive material into the plant liquid effluent requires careful evaluation to determine that hazards are not created. Possible accidental release of excessively high concentrations must be considered.

In some types of industrial or laboratory operations, it may be practical to provide waste containers in the laboratories or shop areas which are to be used exclusively to contain waste radioactive liquids. If the volumes are relatively small and the process does not involve a continuous flow of liquids, the use of individual containers for retention of these waste liquids is practical. If waste liquids are put in individual containers, a periodic pickup procedure must be established, and provisions must be made to dispose of the liquids from some central point. If the

plant also has requirements for disposing of radioactive solids, it may be practical to provide for ultimate disposal of the liquids at the same place that the solids are sent. The ICC Shipping Regulations require that all radioactive liquids be packaged in such a manner that they will be completely absorbed if the container breaks during shipment; this imposes a rather severe restriction whenever such liquids must be shipped from a plant. If, however, the volume is very small, it may be practical to ship liquids to the disposal point. One method that may prove practical in some cases is that of absorbing the liquids in an oil- and water-absorbing material such as the high-absorbent clay which is commonly used in industrial plants for absorbing oil or liquids from floors. The clay is available under various trade names. By absorbing the liquid in this material, the special shipping procedures for liquids are avoided. Depending upon the volume of liquid involved, concentration by evaporation may be desirable. If evaporation processes are used, extreme care must be taken to ensure that the vapor effluent or condensate is not contaminated enough to prevent its release.

For industrial operations involving larger volumes of radioactively contaminated liquid, the handling of individual containers becomes a serious problem. There is always the danger that accidents or spills will cause contamination of an area. The release of these contaminated liquids from the laboratory or manufacturing areas through **drain lines** becomes almost essential. When the amount of contamination is variable, or when accidents may occur which would result in high concentrations of radioactive materials in the liquids, it becomes necessary to monitor these liquids carefully before they are released into storm sewers or other liquid-effluent systems. Therefore, **hold-up tanks** become necessary so that the amount of radioactivity in the liquids can be measured before release. Where the volume of contaminated liquids is less than a few hundred gallons per day and the amount of contamination in the liquid is small enough so that it can normally be dumped into storm sewers, the use of two hold-up tanks, which are alternately filled and monitored, will provide a satisfactory check to ensure against release of excessive amounts of radioactivity. However, provision must be made for disposal of any tanks of liquid which contain more radioactivity than can be released to the regular plant drainage system. If materials with relatively short half-life are involved, an auxiliary storage tank may be used to permit decay of the liquids before release. If short-half-life materials are not involved, however, provision must be made for ultimate disposal which may involve methods of concentration, absorption, or chemical separation.

In any scheme of waste disposal for radioactive liquids, it seems essential that small volumes of liquid, particularly small volumes having large amounts of radioactive materials in them, should always be handled and processed without dilution, unless the normal volume of water available for dilution is always adequate to meet the required tolerances.

When the volumes of contaminated liquid are great, considerable savings can result if two separate drain-line systems are provided. One system, which might be referred to as a **monitor-drain system**, would collect all liquids which inherently have a very low level of radioactivity. This system would also collect from areas where the water is normally not contaminated but where an accident could create high levels of contamination. Connections into this system are made on the basis that the liquids from the complete system will normally be released after they have

been monitored, except in case of an accident. A second drain-line system which might be referred to as a **special monitor-drain system** would be used for collection of liquids which are significantly contaminated from normal operations. Provision must be made to process this water to retain the radioactivity and to release the liquid after it has been sufficiently decontaminated to meet tolerance values (see Sec. 21 for methods of processing liquid wastes).

Even when these two drain systems are used, it is still preferable to retain and process separately any **high-level liquid wastes** of medium or small volumes. Where the volume of high-level wastes is sufficient, it may be desirable and economical to provide separate shielded processing facilities for removal of the radioactivity from these wastes. Drain lines carrying high-level wastes should be constructed as double tubes with the liquid normally carried in the inner tube, and the outer tube being used to detect or collect any leakage from the inner tube. This arrangement will prevent serious underground contamination due to leakage in the pipe carrying the contaminated liquid. By reasonable limitation on the amount of radioactivity in the "special monitor-drain system," it is practical to use standard plumbing arrangements.

NATURAL RADIOACTIVE MATERIALS

Uranium, as obtained from mineral deposits, is the basic raw material for the atomic-energy industry. It occurs in a variety of ores such as carnotite, autonite, uraninite, and pitchblende. It is frequently present as the oxide and may be in either the reduced or oxidized valence state (4 or 6). Wherever uranium is found in nature, the complete chain of daughter products usually is present.

In addition to Uranium-238 and its daughters, all **naturally occurring uranium** also carries with it a small amount of Uranium-235 and all the daughters in this family. Because of the quantities and decay schemes, the Uranium-235 family does not contribute significantly to the health hazards created by these naturally occurring radioactive materials.

When uranium is separated from the ore and from the associated members of these two radioactive families, the resulting mixture of uranium isotopes is referred to as natural uranium. **Natural Uranium** contains the amounts of uranium isotopes shown in Table 11-1, with their specific activities indicated.

Table 11-1. Specific Activities and Half-lives of Uranium Isotopes *

Isotope	Alpha activity in normal uranium/mg/min	Alpha activity of isotope/mg/min	Half-life, years
U ²³⁴	737.4 ± 1.6	(1.345 ± .004) × 10 ⁷	(2.522 ± .008) × 10 ⁵
U ²³⁵	27.2 ± 3.5	3,824 ± 490	(8.8 ± 1.1) × 10 ⁸
U ²³⁸	737.4 ± 1.6	742.7 ± 1.6	(4.49 ± .01) × 10 ⁹
Normal (total of all isotopes)...	1,502.0 ± 1.5		

* From *Phys. Rev.*, December, 1949, p. 1562.

Health hazards in **uranium mining and milling** operations may be caused by external radiation or inhalation of a number of different isotopes. Various daughters of the uranium family may exert a greater influence in some mining or milling operations than in others. Because of this very complicated picture, it usually is necessary to evaluate the potential hazard from each of the isotopes in the family to determine whether one or more members may be the controlling factor at the particular location in question. Where high-grade ores are involved, the potential beta and gamma external radiation exposure may be a very important consideration. For very low grade ores, the self-shielding given by the nonradioactive components of the ore will, in general, reduce the beta-gamma radiation level in mines to a reasonably low level in terms of the permissible exposure level. The same relationships will usually exist in the milling operations.

In uranium-mining operations, the greatest hazard is usually from the daughter **Radon-222** (refer to natural background of radon in air, Sec. 4). Radon, being an inert gas, enters mine atmospheres by diffusion from the ore bodies themselves and from ground water. As the gas diffuses into the mine air, it is free of its decay products, but its immediate daughters have short half-lives so that equilibrium is approached in about 3 hr in still air. Ventilation will prevent this balance, however, and equilibrium conditions are seldom found. Removal of the radon gas from a mine by means of ventilation is one of the effective ways of reducing the concentration of radioactive materials in mine air because it reduces the radon and, at the same time, prevents the build-up of radon daughters. Since under equilibrium conditions in mine air the lung-radiation dose from the daughters is approximately twenty times that from the radon, preventing the build-up of daughters is an extremely important factor in controlling the exposure of persons in this environment.

The evaluation of the potential health hazard from **radon** and its daughter products is usually made by collecting a 100- to 250-liter air sample on filter paper or a membrane and determining the alpha radioactivity on this sample approximately 1 hr after its collection. This reading is then converted to disintegrations per minute and divided by the number of liters sampled to give disintegrations per minute per liter. Reference is then made to Fig. 11-1 to determine the actual concentration in the air at the time of sampling. The counts on the sample can be made at any time within a period of 40 to 90 min after collecting the sample. A factor is obtained from the graph, and disintegrations per minute per liter is divided by the

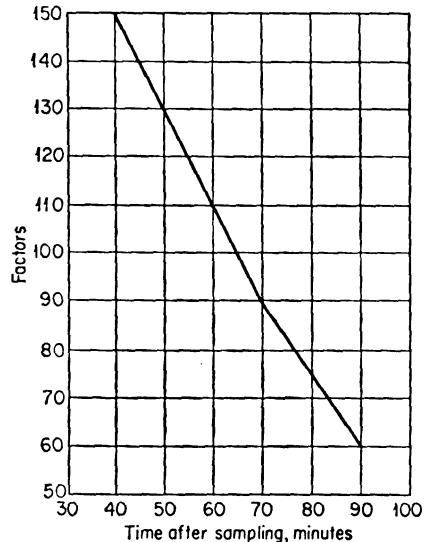


FIG. 11-1. Factors for determining percentage of tolerance level of radon daughter products in the atmosphere. (*Am. Ind. Hyg. Quart.*, March, 1956, p. 87.)

factor to obtain the corrected value. The correction factors obtained from this graph are satisfactory over all ranges of equilibrium ratio. For most conditions, the error is only a few per cent, and the greatest error is 15 per cent on the safe side (see Table 11-2).

Table 11-2. Per Cent Error of Tolerance-level Estimation with Adjusted Factors for Different Times and Equilibrium Ratios *

Equilibrium ratio	Time after Sampling	Factor	% error
1:1:1	40	150	7
	50	130	3
	60	110	3
	70	90	3
	80	75	7
	90	60	2
1:0.9:0.8	40	150	8
	50	130	4
	60	110	3
	70	90	12
	80	75	10
	90	60	4
1:0.45:0.35	40	150	2
	50	130	2
	60	110	2
	70	90	4
	80	75	11
	90	60	6
1:0.15:0.06	40	150	7 †
	50	130	7 †
	60	110	2 †
	70	90	3
	80	75	12
	90	60	7

* From *Am. Ind. Hyg. Quart.*, March, 1956, p. 87.

† % underestimated.

It appears that the determination of **Polonium-210** in urine can be used as an index of exposure to radon and its decay products (Sultzzer, M., and J. B. Hursh, Polonium in Urine of Miners Exposed to Radon, *A.M.A. Arch. Ind. Health*, vol. 9, p. 89, 1954). Considerably more experience will be needed before this method can be quantitatively applied.

In uranium-mining operations, the dust produced will contain uranium, radium, and the daughters of radon. The theoretical radium to uranium ratio in ores is approximately 1 part by weight of radium to 3 million parts by weight of uranium. Depending upon the type of ore, this ratio may be as high as 1 part of radium to 2 million parts of uranium. The specific activity of radium is approximately 3 million

times that of uranium, and its permissible concentration in air on a radioactivity basis is approximately half that of uranium. The dust will therefore contain a significant amount of both radium and uranium, and the exact contribution to the hazard by each will depend upon the ratio in the ore. The concentration of these elements in the air can be determined by collecting dust samples on filter papers or membranes and allowing all the short half-life daughters to decay for a period of at least 24 hr. The same samples collected for radon decay products can be used. Alpha counts on these samples will then give data for calculating the long half-life elements, which will be mainly uranium and radium.

Milling operations usually involve the crushing of the ore to approximately 60-mesh size. This usually is done in two steps: first, by a very coarse crushing and, second, by both crushing and milling in bar mills or rod mills. The milled material

Table 11-3. Summary of Potential Hazards from Processing Pitchblende Containing 25 Per Cent Uranium *

Potential hazard	Source of exposure	Potential exposure under normal operating conditions but with no controls	Remarks
Gamma radiation	Handling ore	50 mr/hr adjacent to stack of drums of ore	Radium content of ore approximately 100 mg/ton
	Precipitates containing radium	100 mr/hr adjacent to stack of drums	Ra ²²⁶ content approximately 300 mg/ton
Beta radiation	Conversion of UF ₄ to UF ₆	2-3 rad/week possible among workers loading and unloading reaction vessels; dose to hands may be higher	Gaseous UF ₆ is removed, leaving unreacted residue highly enriched UX ₁ , UX ₂
	Vacuum-casting uranium billets	Furnace operators may receive as much as 2 rad/week; dose to hands somewhat higher	Sources of beta radiation are distilled impurities which contain UX ₁ , UX ₂ and condense on furnace interior
	Handling metallic uranium	Dose to basal epithelium of skin approx 235 mrad/hr when in contact with U metal in equilibrium with UX ₁ , UX ₂	90% of equilibrium beta activity is restored 90 days after vacuum casting
Radon inhalation	Storage of ores in closed spaces	With no ventilation concentration may range as high as 10 ⁻⁸ curie/liter	Conditions much improved by even minimal ventilation in storage and processing areas
Inhalation of alpha-emitting dusts and fumes	Sampling and crushing ore	Daily average concentrations up to 5 mg/m ³ cubic meter have been experienced at operations in which good process and area ventilation was not provided	Dust-control techniques normally employed in processing toxic metals are effective
	Mechanical or manual handling of dry U salts and oxides	Daily average concentrations up to 5 mg/m ³ have been experienced at operations in which good process and area ventilation was not provided	Dust-control techniques normally employed in processing toxic metals are effective
	Escape of gaseous UF ₆	Massive exposures due to sudden releases of UF ₆ may occur	High standards of maintenance are required to prevent leaks at valves, flanges, etc.
	Machining metallic U	Excessive exposures possible from pyrophoric nature of U	Store chips under oil; use liberal quantities of cooling oils

* From A.M.A. Arch. Ind. Health, July, 1956.

is then leached with carbonates or acids which dissolve out the uranium. Other metal separations such as vanadium may also be involved. The liquids containing the uranium are treated to remove the uranium, by either chemical processing or ion exchange, and the concentrated uranium oxide is placed in drums for shipment to processing plants. In the milling operations, radon and its decay products are not usually an important factor contributing to potential health hazards. Dusts from the crushing process will contain uranium, radium, and polonium, as well as other potentially hazardous materials such as silica. These hazards are evaluated by collection of dust samples from the breathing zone of workmen and determining the concentration of long-half-life radioactive materials by standard methods. The hazard from silica or silicates is evaluated by dust-counting procedures commonly used in ordinary industrial-hygiene work. The usual milling operation involves the processing and handling of ores containing, on the average, only a few tenths of a per cent of uranium, and the external radiation exposure from beta-gamma emission is not great in terms of the permissible levels. Only under special conditions where processing of high-percentage uranium ores might be involved would a higher external radiation be involved.

The concentrate from the mills and, in some cases, very high grade ores are treated in uranium-processing plants to remove all impurities and produce a high grade of uranium suitable for a variety of purposes in atomic-energy work. The **purified uranium** may be in the form of metal, oxide, fluoride, or occasionally as some other salt of uranium. The processing of ores is done in a variety of ways, depending upon the type of ore and concentration of uranium. Table 11-3 gives a summary of potential hazards from processing pitchblende containing 25 per cent uranium.

REACTOR-CORE MANUFACTURING

Reactor-core manufacturing may involve a variety of metallurgical, machining, chemical treatment, and material-handling operations. The raw materials may be metallic natural uranium, metallic uranium enriched in the 235 isotope, natural uranium oxide, enriched uranium oxide, and a wide variety of nonradioactive metals such as aluminum, zirconium, boron, beryllium, and stainless steel. Thorium and plutonium, in either metal or oxide form, may also be used.

During the manufacturing operation, various potential health hazards exist. These include the inhalation of dusts and fumes, external radiation exposure, contact with surface contamination, and the possibility of fire or explosion.

The major potential health hazards from the radioactive materials used in reactor-core manufacturing are **dusts and fumes from uranium**. If thorium is involved, see page 11-50. The maximum permissible concentration of air-borne uranium breathed by industrial workers is 5×10^{-11} microcurie/ml of air. This is equal to 110 alpha disintegration/min/m³ of air. (Reference should always be made to the latest revision of NBS Handbook 52.) For natural uranium this is equal to approximately 75 μg of uranium/m³ of air. Where **uranium enriched in the isotope 235** is used, the permissible concentration in terms of microcuries per milliliter of air and alpha disintegrations per minute per cubic meter of air can be considered to be the same as for natural uranium. The weight concentration in air, however, will be much less because of the higher specific activity of the enriched uranium. The en-

riched uranium usually contains a significant amount of Uranium-234 as well as a higher percentage of Uranium-235, and the resulting specific activity may be about 100 times that of natural uranium. For this ratio, the permissible concentration of enriched uranium in air would be approximately $0.75 \mu\text{g}/\text{m}^3$ of air. It becomes evident that, although the maximum permissible concentration for both natural and enriched uranium in air is assumed to be the same in terms of radioactivity per unit volume of air, the problem of adequately controlling dust or fume hazards when working with enriched uranium is much greater than the problems involved in controlling such factors for natural uranium.

There is a limited possibility of **external radiation exposure from natural uranium**, but the **enriched uranium** presents no such problem. Large quantities of natural uranium give off sufficient beta and gamma radiation to require an evaluation of this potential exposure. As a rule of thumb, the external radiation exposure should be checked where personnel work in close proximity to more than 5 lb of natural uranium. Preliminary estimates of personnel exposure can be made with survey meters. The actual exposure of personnel can be checked by means of film badges. Pocket chambers or self-reading pocket dosimeters must be used with discretion because they do not record the low-energy beta radiation. Radiation dosages in excess of 100 mrem/week are uncommon unless a person is spending a considerable amount of time in close proximity to large pieces or quantities of natural uranium, or near an unshielded storage location. Since a considerable percentage of the dose may be coming from beta radiation, sufficient shielding to stop the betas will significantly reduce the external-radiation-exposure level. A $\frac{1}{8}$ -in. sheet of iron or its equivalent is sufficient for this purpose. After suitable evaluation, the continuation of personnel monitoring may not be necessary.

At the present time there are no generally accepted permissible levels for radioactive **surface contamination** (see also Sec. 18). It is desirable to keep the uranium surface contamination in the uranium-processing area as low as is practical to minimize the radiation exposure of personnel, to reduce the possibility of contamination of persons and personal clothing, to aid in preventing the spread of uranium contamination beyond the processing area, and to reduce the time and expense required to decontaminate the area if the uranium work is discontinued. The control level for uranium surface contamination should be determined by the type and quantity of uranium work to be done, the degree of segregation of the uranium-processing area from adjacent plant areas, the ease with which the area can be decontaminated, the consequence of uranium contamination in other products and operations in the plant, and the future use to be made of the uranium-processing area. There may be cases where surface contamination or inclusions in the surface of an end product will dictate careful contamination control. It is usually practical to differentiate between fixed contamination which cannot be removed by simple cleaning methods and removable contamination. Fixed contamination will contribute only to external radiation exposures unless the material containing it is physically or chemically altered. Smear tests will measure the amount of removable contamination, whereas survey meters are used to detect and measure that which has become fixed in the surfaces. As a guide, it is sometimes suggested that uranium contamination on surfaces not normally in direct contact with uranium in the processing area should not exceed an average of 100 alpha and/or 2,000 beta-gamma dis-

integrations per minute per square foot of surface area as measured by the smear test. This does not apply to machine-tool work surfaces such as rolling-mill rolls and extrusion dies. Surface contamination can be higher than this without directly causing excessive concentrations in the air (Eisenbud, Blatz, and Barry, How Important Is Surface Contamination? *Nucleonics*, vol. 12, pp. 12-15, August, 1954). Bulk contamination such as uranium scale should be cleaned up as soon as practical after it is produced to prevent the spread of contamination in the area. It is usually practical to keep such materials off the floors or work surfaces by proper design of the processing equipment or temporary use of blotting paper or trays. Controls must be established to ensure that any equipment or materials leaving the plant are not significantly contaminated with uranium. These contamination limits should be based on regulations such as those of the ICC (see Sec. 3), state regulations, or limits imposed by subsequent handling requirements or uses of the material. Surfaces whose smear counts are less than 10 alpha and/or less than 200 beta-gamma disintegrations per minute per square foot of surface area can be considered clean. Machine tools should be decontaminated before being used on materials other than uranium. No significant uranium contamination should be permitted to spread beyond the uranium-processing area.

Uranium is a **pyrophoric metal** similar to zirconium, magnesium, and titanium. Large pieces normally do not burn without the continued application of external heat. However, chips, turnings, and fines ignite easily (sometimes spontaneously) and burn with an intense heat. This intense heat can cause spalling of concrete and distortion of machine-tool components such as lathe and milling-machine beds. Significant amounts of uranium oxide fumes are produced by burning uranium. Only dry sand, Met-L-X powder, or G-1 powder can be used in the extinguishing of **uranium fires**. Water or other materials containing oxygen will increase the burning rate if used on a uranium fire. The treatment of **uranium-zirconium alloys** containing 1 to 50 per cent zirconium with nitric acid media may result in an explosion. Similar incidents may occur with other uranium alloys. Any coating or film produced during pickling of uranium alloys is potentially explosive (Larson, Shor, Feder, and Flikkema, A Study of the Explosive Properties of Uranium-zirconium Alloys, *Argonne Natl. Lab. Rept.* ANL-5135, July, 1954).

The **dust concentration** in the uranium-processing area should be checked periodically to ensure that the health of personnel in the area is not endangered. A small portable air sampler should be available for taking air samples. For more accurate evaluation, aerosol sampling equipment using membrane filters is recommended. Dust samplers employing electrostatic precipitation can be used. Qualitative determination of the transferable surface contamination inside and outside the uranium-processing area may be made by using the smear or swipe tests. Techniques for such tests are described in Sec. 18. After analysis, the result is usually expressed as disintegrations per minute per unit of area, the unit of area being either 1 sq ft or 100 sq cm. A portable survey meter, particularly one provided with a thin window, is useful for detecting high levels of contamination and for evaluating potential exposures to external sources of radiation. The common survey instruments not provided with special thin windows will not detect alpha radiation and hence are relatively insensitive for detecting enriched uranium.

Air samples, smear samples, evaporated liquid samples, etc., can be analyzed by

counting the radioactivity emitted from the samples in suitable electronic equipment (see Sec. 20). An internal-type gas-flow proportional counter and scaler is recommended for a small installation since it will detect both alpha and beta radiation, and has adequate sensitivity for a variety of samples. Where only natural uranium is involved, it might be practical to determine the uranium in these samples by a chemical analysis.

Preventive measures to protect the health of personnel engaged in uranium-manufacturing processes include those usually associated with the handling of radioactive materials—the use of protective clothing, the observance of good-housekeeping rules, and the exercise of proper precautions in storage and shipment. In addition, provisions often must be made for local ventilation and for prevention of fire or explosion.

Uranium-machining operations should be performed so as to prevent the chips and turnings from being ignited by the heat produced by the cutting operation. This can be accomplished by using an adequate supply of a high-flash-point mineral oil or water-soluble coolant, a sharp cutting tool, proper cutting speed, and proper depth of cut. Chips and turnings should not be permitted to accumulate where they can be ignited by a hot chip or turning from the cutting tool. **The storage of uranium chips, turnings, and fines** should be done under at least 2 in. of an inexpensive high-flash-point mineral oil. The containers should be vented to release any gaseous hydrogen which is slowly evolved. Hydrogen is produced more rapidly if the uranium is in contact with water. Metallic uranium dusts are explosive when suspended in air in the proper concentration. Uranium oxide is not flammable, and uranium dioxide (UO_2) is not flammable at ordinary temperatures. Dust clouds of uranium oxide in air cannot be ignited at temperatures up to 1000°C .

Local exhaust hoods are often required as a necessary part of a machine tool to prevent breathing hazards and the spread of contamination while working with uranium. A dust collector is usually required as a part of the exhaust system to prevent dispersion of the dust or fumes outside the building. From the viewpoint of surface-contamination control and dust collection, it is generally advisable to collect the dust or fumes at the source of generation rather than to rely on general ventilation to control the breathing hazards. Operations which are normally considered to produce insignificant quantities of dust can produce hazardous concentrations of air-borne uranium.

Uranium-machining or processing areas should be separated and restricted from other operations in the plant. The degree of separation and restriction should be a function of the type of work to be done, the effectiveness of exhaust systems, and the operating techniques of the personnel. If a separate room or walled-off area is available, this is often the best solution to this problem. The area should be adequately identified as a radiation or radioactive hazard area.

Walls and floors of the area may be made nonporous and smooth to reduce clean-up time and to make it possible to return the area quickly to other use after the uranium work is completed.

Protective clothing should be provided for employees or visitors in the area where uranium is processed if there is a likelihood that personal clothing may become contaminated. The type and extent of the protective clothing could range from lab coats and shoe covers to a complete fitting of shoes, socks, underwear, coveralls,

gloves, and head covering. Personnel traffic should be restricted to an entrance where the protective clothing is provided. The protective clothing should be put on when entering the area and removed when leaving. Water from laundering the radioactively contaminated clothing should be checked for uranium contamination and disposed of in a safe manner. This may require laundering facilities at the plant or the services of a licensed commercial laundry adequately equipped to process radioactively contaminated clothing.

The type of work, the length of time the work is to be performed, and the amount of contamination in the area may make it necessary to provide a complete change room with lockers and showers and one or more radiation-measuring instruments for checking contamination.

All uranium should be properly identified as radioactive material to avoid misplacement which could result in health hazards.

The **shipment of radioactive material** must be done in accordance with ICC Regulations (see Sec. 3). The **shipment of enriched uranium** must be done in such a way that a **critical mass** cannot be assembled during shipment. Whenever enriched uranium is stored or is being processed, the quantity in any one location must be carefully controlled to prevent the possibility of assembling a critical mass. This requires expert analysis and rigidly controlled handling procedures. Special storage vaults may be required. See Sec. 15 for further information.

Arrangements must be made for the **disposal of wastes** of any nonaccountable solid or liquid materials in such a manner that they will not subsequently create a problem. Reasonable advantage can be taken of environmental factors such as dilution and dispersion to minimize the cost of disposal. Retention tanks to permit analysis of contaminated liquids before release from the plant, and treatment facilities for evaporation, filtration, and ion exchange of highly contaminated liquids may be required. **Metal scrap** which has been surface-contaminated by uranium can be released to commercial scrap channels if the level of radioactivity is not too high. The following requirements have been established by the Atomic Energy Commission for their operations.

DISPOSAL CRITERIA

(From AEC Manual Chapter 5182)

Before disposal, all scrap proposed for sale shall be monitored using appropriate instruments and techniques, by qualified personnel. No scrap shall be sold having an average alpha activity greater than 5,000 disintegrations per 100 square centimeters of surface or a peak alpha activity greater than 25,000 disintegrations per minute measured over 100 square centimeters of surface. The person monitoring the scrap shall sign and file a report stating his readings and observations and describing the lot of scrap involved. The scrap to be disposed of shall not include the following:

- a. Materials having readily removable loose contamination; and
- b. Containers, pipes, tubing or machinery in which gross quantities of uranium-bearing materials can be entrapped and which cannot be checked by suitable monitoring techniques.

Burnable scrap, contaminated with small amounts of uranium, can be burned in the open without causing significant contamination problems if the ash is collected and handled as a contaminated material (Harris and Weinstein, Open Field Burn-

ing of Low Level Contaminated Combustible Wastes, *Am. Ind. Hyg. Assoc. Quart.*, vol. 17, no. 4, pp. 388-390, December, 1956). Burnable scrap containing large amounts of uranium may require a special incinerator with a flue-gas cleaner.

A physical examination is advisable for employees who are to work with uranium to detect abnormal conditions which might be aggravated by exposure to uranium and/or radioactivity, or which may mistakenly be attributed to exposure. Where persons might breathe or ingest significant amounts of uranium over a period of months, periodic analysis of urine samples for uranium will be helpful in evaluating and controlling their true exposures.

REACTOR OPERATION AND MATERIALS

Reactor classifications are used to designate industrial nuclear reactors which serve one or more of five different purposes. The attendant hazards of their operation are quite different. Experimental reactors, which include critical assemblies, are of a flexible design which allows rapid alteration of the core components to stimulate varying operating parameters. Research reactors are used in pure and applied nuclear research, and are usually intended to supply a high flux of neutrons. Breeders and converters provide a high neutron flux for converting Uranium-238 to Plutonium-239, and Thorium-232 to Uranium-233, the produced isotopes being fissionable. Test reactors are used to study materials under high neutron and gamma fluxes, the data obtained being used to develop improved materials for reactor designs. Power reactors are designed to extract the thermal energy released in fission and make it available for practical use.

Nuclear-reactor fuels include both natural and enriched uranium. Only one of the isotopes of natural uranium, Uranium-235 (relative abundance 0.71 per cent), is fissionable, and the important parameter in reactor design is the amount of Uranium-235 present in the core, rather than the total amount of uranium. Although the original reactors were designed to use natural uranium as a fuel, higher neutron fluxes and more compact reactors are now available through the use of enriched uranium. Enriched uranium has been processed to increase its percentage of Uranium-235. Uranium-233 and Plutonium-239 are available in the nearly pure state, and the use of these fuels results in the smallest reactor-plant size. Uranium-235 is presently the most widely used nuclear fuel, and the discussion presented here will concentrate upon reactors using Uranium-235, although extrapolations to other fuels may be made easily.

The neutron energy produced in fission in the **thermal reactor** is reduced by collisions of the neutrons with moderating atoms, in order to increase the probability of subsequent fission reactions in the fuel. **Fast reactors** operate with fission-energy neutrons and require no moderator. **Intermediate reactors** use neutrons of energy between the above two extremes.

The **moderator** in a thermal or intermediate reactor is used to slow the fission-energy neutrons to the reactor operating energy. A good moderator has two attributes: its nuclear mass is nearly the same as the mass of a neutron, and it has a low neutron-absorption cross section. A pure **carbon moderator** is often used in the research-type reactor. It is easily machined and of very low toxicity. Carbon has a low neutron-activation cross section so that it does not present a radiation hazard

after use in a reactor. Water, both heavy and light, is employed in the dual capacity of moderator and energy-transfer medium in many power reactors.

On the basis of its nuclear and thermodynamic properties, **beryllium** is an excellent moderator. However, the relatively high toxicity and the machining difficulties associated with the processing of beryllium present serious problems. The maximum allowable concentration for beryllium is generally accepted as $2\mu\text{g}/\text{m}^3$ of air (*AMA Arch. Ind. Health*, vol. 14, p. 189, 1956). Some users of beryllium accept a short-term exposure value of $25\mu\text{g}/\text{m}^3$. The prospective user of beryllium should refer to the current literature on the subject as well as to the volume "Pneumoconiosis" [Vorwald, A. J. (ed.), Harper, New York, 1950] and "M & R Beryllium Machining History" (Case, Watkins, and Jones, Y-868, Technical Information Service, Oak Ridge, Tenn., 1952). Organic materials, which contain large amounts of carbon and hydrogen, are excellent moderators, although they are subject to radiation damage.

Reactors which have the fuel and moderator intermixed as a solution or slurry are termed **homogeneous reactors**. When the fuel occurs as discrete elements or bundles, the reactor is termed a **heterogeneous reactor**. The homogeneous-type reactor facilitates the continuous reprocessing of fuel during reactor operations, giving rise to fission-product handling. The fuel from a heterogeneous reactor must be removed from the reactor vessel and reprocessed in a separate plant facility so that the hazards of reprocessing are not directly associated with the reactor-plant operation.

Mention has previously been made of the use of water as a coolant. Air may be used as a coolant, usually when the reactor is not designed for power production. Certain gases, such as carbon dioxide, are being used as coolants. Some liquid metals and eutectics (bismuth, lead, sodium, potassium, mercury) have favorable thermodynamic properties and are used as reactor coolants.

The **fission process** in the nuclear core gives rise to three important products: neutrons, thermal energy, and fission products. The neutrons are required to continue the nuclear-fission process in the core. They will be considered more fully below. Although a commercially important item, the thermal energy is frequently not used and must be dissipated in a reliable fashion. Each fission which takes place in the nuclear core results in the release of 3.2×10^{-11} watt-sec of energy, of which 87 per cent is immediately available and 5 per cent will be dissipated over a long period of time as the fission products undergo beta decay (the balance of fission energy is carried off by neutrons and is not available for use). The 5 per cent of the total fission energy given off by the fission products necessitates continuous use of the core-cooling system even after the nuclear core has been shut down. Failure to dissipate this energy may result in a melting of the nuclear-core components with a subsequent release of the accumulated fission products to the environment. The rate of release of fission-product decay energy will decrease with time in accordance with the total-fission-product-decay equation (see below).

Fissionable nuclei (U-233, U-235, Pu-239) are neutron-rich, and not only are free neutrons emitted by the fission process, but the fission products are also neutron-rich. There are some 65 identified **fission products** ranging in mass number from 72 to 158. Nearly all these fission products are radioactive and decay by beta and gamma emission, each individual fission product decaying with its own charac-

teristic half-life. Not only are the fission products themselves radioactive, but the daughter nuclei are usually also radioactive. Indeed, long fission-product decay chains are established. For example:

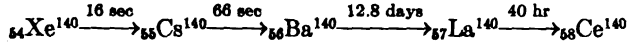


Figure 11-2 shows the relative contribution to the remaining fission-product activity of the various fission-product isotopes and their decay daughters for a de-

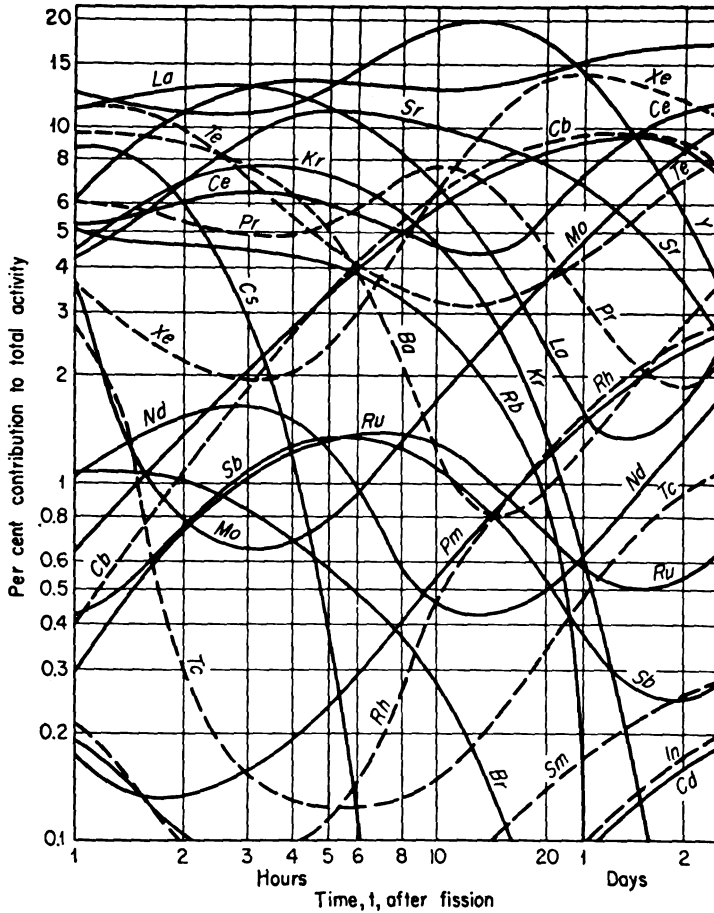


FIG. 11-2. Individual and total rates of decay of the fission products. (Hunter and Ballou, ADC-65, p. 73, Fig. 2.)

cay time to 100 years. Figure 11-3 shows the relative contribution to the remaining fission-product activity of the various fission-product elements for a decay time to 1,000 days. These graphs are reproduced from "Simultaneous Slow Neutron Fission of U-235" (Hunter and Ballou, "Individual and Total Rates of Decay of the Fission Products," ADC-65, Technical Information Service, Oak Ridge, Tenn., 1949).

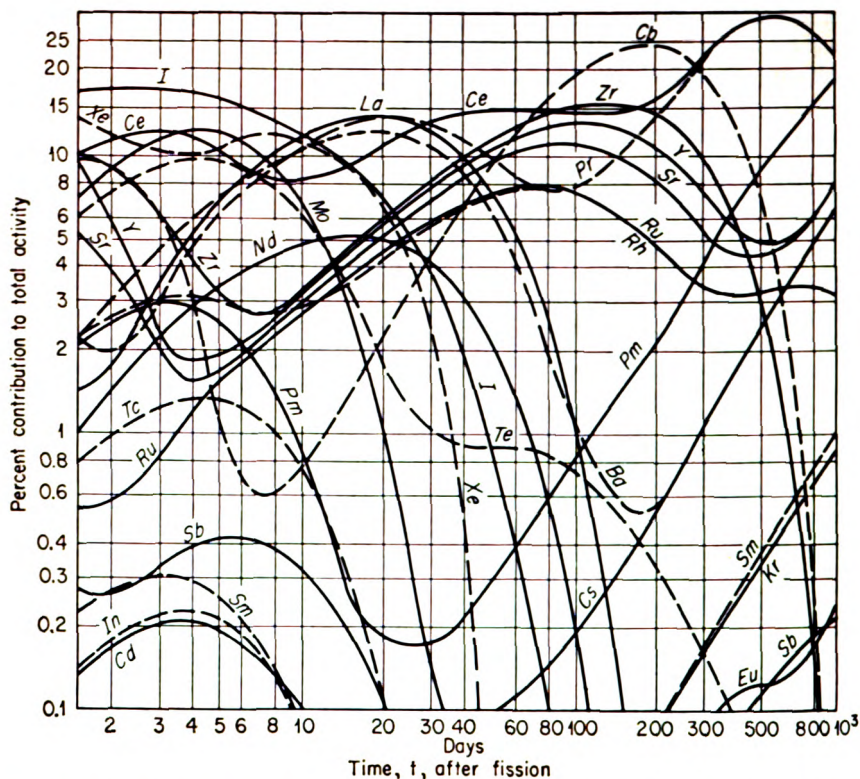


FIG. 11-3. Individual and total rates of decay of the fission products. (Hunter and Bal-lou, *ADC-65*, pp. 77, 78, Fig. 3.)

On the average, each fission product is the precursor of two other radioactive nuclei. An approximate representation of the over-all rate of **energy emission** from each fission is given by

$$\text{Rate of emission of gamma-ray energy} = 1.26t^{-1.2} \text{ Mev/sec}$$

$$\text{Rate of emission of beta plus gamma energy} = 2.66t^{-1.2} \text{ Mev/sec}$$

As an example of the use of the above equation, let us assume that a new reactor core has been operating at a steady power state of 3 megawatts for 30 days. The core is then shut down. It is desired to find the energy release rate 12 min following shutdown. The fission rate inside the operating core would be

$$\frac{3 \times 10^6 \text{ watts}}{3.2 \times 10^{11} \text{ watt-sec/fission}} = 10^{17} \text{ fission/sec}$$

The elapsed operating time is

$$t = 30 \text{ days} \times 24 \text{ hr/day} \times 60 \text{ min/hr} \times 60 \text{ sec/min} = 2.6 \times 10^6 \text{ sec}$$

The elapsed shutdown time is

$$t_1 = 12 \text{ min} \times 60 \text{ sec/min} = 720 \text{ sec}$$

Then

$$\begin{aligned} E(\beta + \gamma) &= 2.66 \times 10^{17} \int_{2.6 \times 10^6}^0 (t_1 - t)^{-1.2} dt \\ &= \frac{2.66 \times 10^{17}}{0.2} (t_1 - t)^{-0.2} \Big|_{2.6 \times 10^6}^0 \\ &= 3.2 \times 10^{17} \text{ Mev/sec} \end{aligned}$$

The rate of fission-product energy release 12 min following shutdown will be 3.2×10^{17} Mev/sec or 5.1×10^5 watts. An insight into the total number of decay processes occurring in the core may be gained from the fact that the average fission-product beta particle carries away 0.4 Mev and the average fission-product gamma ray carries away 0.7 Mev of energy.

The **reactor-coolant system** of a nonpower reactor is frequently designed to use air as a coolant. In spite of its low heat capacity and thermal conductivity, it is economical to use air by virtue of its abundance. It should be noted that air contains 0.92 per cent Argon-40, which is activated by neutrons to **Argon-41** ($T_{1/2} = 1.8$ hr, beta 1.25, 1.55 Mev, gamma 1.3 Mev). A medium-power air-cooled reactor will produce 20 to 40 curies/day of Argon-41. During unstable weather conditions (inversion), the Argon-41 may lie in the vicinity of the reactor stack and become an external radiation problem to the reactor personnel and local inhabitants.

An ion-chamber-type instrument having a low range of detection is used for **detection of Argon-41**. As argon does not enter into any physiological processes, the radiation exposure from it is evaluated as an external dose. Exposures can be calculated by assuming the body to be immersed in a hemisphere of radioactive isotope. One air-cooled nuclear-reactor installation (Brookhaven) has used a series of outlying monitor stations with recording ionization chambers to appraise the Argon-41 problem.

In order to detect abnormal quantities of radioactivity from the core which may indicate burst fuel cartridges, monitors are placed in the cooling-air ducts beyond the reactor core. High-efficiency filters will remove solid fission products and prevent the release of gross activity from burst fuel cartridges to the environment.

A single pass is generally used when **water** is the **coolant** in nonpower reactors. Sources of water for nonpower reactors are clean rivers and wells. Prior to passing through the water channels in the core, the water must be filtered for particles that might clog channels and cause "hot channels" (see Major Reactor Accidents on page 11-34). Also, dissolved minerals must be kept to a minimum to prevent poisoning of the core, and to prevent the activation of a large amount of dissolved minerals which would prevent rapid disposal of the water following its passage through the core. As with any industrial use of large quantities of cooling water, the temperature of the water should be returned to near normal to prevent upsetting the environment. Retention basins are used to allow time for cooling and for decay of dissolved activated isotopes.

Water used as a coolant for a power reactor presents a somewhat different picture. Here the water is used in a closed cycle and constantly recirculated. Prior to use

the water is filtered, deionized, and deaerated. In spite of the care taken to ensure clean water, the cooling-water contamination will normally result from three major sources: (1) rupture of fuel elements discharging fission products into the water; (2) activation of scale swept through the nuclear core, particularly cobalt-manganese from stainless steel; and (3) activation of dissolved Argon-40 in the water. The first two sources of contamination are kept under control by constantly passing a portion of the cooling water through filters and ion-exchange resins. In spite of the constant recirculation, there will be a build-up of contamination on all inner surfaces of the cooling system resulting from redeposition of the scale. In some cases, the deposited material may become an external radiation hazard, and shielding may be necessary to protect personnel in the area. The noble-gas fission products, krypton and xenon, will not be removed by the filters, and leakage or drainage of the coolant will release these gases. Since they have radioactive daughters, both the noble gases and their daughters may be present in the air.

Activation of the dissolved argon in water is of concern only when proper ventilation cannot be provided for the coolant-loop compartments. The dissolved argon in the water (0.38 cc argon per kg H_2O at $70^\circ F$ and 1 atm) will become activated to Argon-41 in the core of the reactor, and any escape of the coolant, either directly or through the secondary cooling system, may cause an exposure problem.

For maintenance purposes, any portion of the reactor-cooling system should be approached as a contamination and radiation hazard until proved to be otherwise. Even when a secondary system is employed in the power-generating system, the secondary loop may become contaminated from leaks in the primary boiler. The area in which work is to be performed should, therefore, be blocked to traffic, and all persons entering the area should wear protective clothing, which is removed at the time of egress.

Covering the floor with blotting paper will help considerably in the control of contamination, and the provision of local exhaust ventilation when the cooling system is opened is essential, although not a guarantee that respirators or air masks are not necessary in the area. Any tools used on the cooling systems should be considered contaminated until they are decontaminated. No contaminated tools, equipment, or clothing should be taken from the area without being decontaminated or sealed in waxed or plastic bags. Thorough monitoring for air and surface contamination and for radiation should be carried out whenever the cooling system is opened to check the spread of contamination and the possibility of overexposure of personnel.

It may be necessary to perform some **decontamination** of portions of the coolant system prior to starting replacement work. Dilute solutions of nitric acid or Versene have proved useful for this purpose. Care should be taken that these solutions are not allowed into resin-exchange columns or filter media where they would reentrain all collected activity.

Although **liquid-metal reactor coolants** pose serious radiation problems because of their activation, the greatest problem is protection from chemical reactions. Sodium and potassium both undergo violent reaction with water or air, and special provision must be made to cope with this fire hazard. The corrosion difficulties associated with all liquid metals present difficult metallurgical and engineering problems.

Shielding for radiation sources, particularly where neutrons are involved, is an important problem of reactor design. Any new shield should be carefully checked by means of sensitive gamma and neutron meters and film. For new reactor installations the use of artificial sources (cobalt and polonium-beryllium) during the construction stage is helpful in determining the presence of voids. Following the completion of the nuclear installation, a thorough radiation monitoring of all areas should be carried out to ascertain that no radiation hazards exist. Particular attention should be paid to detecting small beams.

It should be noted that the provision of shielding to reduce the radiation level to a "weekly permissible dose rate" or below is not essential for all areas. Many areas around the nuclear installation will require only **limited access** or access only when the reactor is subcritical. In these areas the effectiveness of the shield may be reduced, with consequent economy of design. There are two points which must be considered when partial shielding is used. No neutron activation of structural or component parts should take place in accessible areas. Gamma and neutron radiation both undergo scatter, so that, even though a shield may be effective for the line-of-sight path, scattered radiation may present a hazard to occupants of nearby areas.

Radioactive oxygen in the cooling water which passes through the main reactor shield poses a radiation problem. The **Oxygen-16** in water reacts with neutrons to form **Nitrogen-16** which, on decay, emits a 7-Mev gamma ray with a 7-sec half-life. Other reactions, of less importance, which will take place in the cooling water are $O-17(n,p)N-17$ and $O-18(n,\gamma)O-19$. The Nitrogen-17 formed decays by beta decay to excited Oxygen-17 ($T_{1/2} = 4.1$ sec), which emits a $0.92 =$ Mev neutron to form Oxygen-17. The O-19 decays by beta emission ($T_{1/2} = 29$ sec) and emits a 1.6-Mev gamma ray. Therefore, primary cooling-water lines outside the main reactor shield require shielding of their own. Any withdrawal of the coolant should be through a system which allows adequate decay time for the nitrogen and oxygen isotopes.

The "Reactor Shielding Design Manual" (TID-7004, Office of Technical Services, Department of Commerce, Washington 25, D.C.) contains considerable data important to the reactor-shield designer.

Contamination control in reactors will, in general, involve surveillance in two distinct sections. One of these will consist of the offices, maintenance shops, utility areas, and possibly the control room where no contamination is to be tolerated at any time. The remaining areas, such as the reactor room, canal area, and fuel-loading areas, will frequently be subjected to contamination, but they should be surveyed frequently and decontaminated as necessary. The control of contamination consists of more than mere containment; it is an endless job of checking all areas for sources of contamination and monitoring all equipment being removed from the contamination-suspect area. Taking routine smear tests of all areas is an acceptable method of statistically checking for loose contamination. The frequency of the routine check will depend much on the individual circumstances; an average frequency would call for weekly checks in all clean areas and daily checks in all contaminated areas. Over and above the routine checks, there will be numerous cases where special checks of individual areas must be made. Routine and nonroutine air samples may be used to ensure the absence of aerosol contamination.

The provision of adequate **decontamination facilities** for both materials and personnel is important to the control of contamination. Normally, convenient location of probes and hand and foot monitors to be used when egressing from the contaminated rooms will prevent contamination of clean areas. A sink and decontamination supplies for scrubbing should be located convenient to each monitor station. Probe-type instruments, both alpha and beta-gamma, may be used for checking clothing and equipment for gross contamination.

For the small nuclear-reactor installation, laundering of contaminated clothing will be a problem. Contaminated clothing cannot be sent to commercial laundries unless these laundries are willing to set aside machines for use only on contaminated clothing, and to check the activity of the water for compliance with pollution codes prior to release. With a small operating crew and low volumes of contaminated clothing, domestic-style laundry machines may be used, the effluent water being treated, if necessary, with other reactor liquid wastes.

Disposal of radioactive wastes is a problem which must be met by every nuclear-reactor establishment. In general, there will not be sufficient quantities of solid and liquid disposable wastes from a single nuclear reactor to warrant providing facilities for ultimate disposal. There are a number of private organizations that will contract for such ultimate disposal of the waste. Provision must be made, however, for storage and preliminary packaging.

Larger installations will have to evaluate the economics of contract disposal vs. their own processing plant. Ultimate disposal locations will depend on government regulation. **Combustible contaminated wastes** may be burned in incinerators which have stack-gas cleaners on them. In practice, great difficulty will be encountered in obtaining complete combustion, and a considerable amount of tars and resins will go out the incinerator flue and plug the stack-cleaning system prematurely. The use of wet collectors will remove some of these tars and also will reduce the stack-gas temperature to a level that will not damage the ultimate stack-gas-cleaning system. For low-level wastes, the wet collector may provide sufficient cleaning of the exhaust air. The ultimate stack-gas-cleaning system may take the form of fibrous media, as rock-wool batts, or may be of the electrostatic-precipitator type. In charging the incinerator, care should be taken not to include contaminated materials such as Carbon-14, which will be converted to a noncollectible gas.

Noncombustible wastes are often disposed of by packaging in 55-gal drums with cement shielding to a weight of 75 lb/cu ft. These drums are then dropped into the ocean. In packaging material for disposal in this fashion, care should be taken to keep the radiation from the drum inside the ICC limits, or special arrangements must be made to ship the material (see Sec. 3).

The disposal of radioactive liquids may be accomplished by incorporating them into concrete for sea disposal in drums, or if only small quantities are present, they may be siphoned into the plant sewage system in such a fashion that the plant effluent never exceeds the existing codes for radioactive content. Large volumes of liquid radioactive wastes which do not contain appreciable quantities of soaps or chemicals may be concentrated by the use of ion-exchange-resin columns. When large volumes of wastes of high activity are to be handled, distillation may be used to achieve concentration prior to packaging for disposal.

Radioactive aerosols are normally collected on high-efficiency filters and the effluent air discharged at a point above the highest roof-line area (see Sec. 21).

Irradiated-source handling refers to the equipment and operating procedures for irradiating small quantities of material in a nuclear reactor. The materials are usually placed in sealed aluminum or nylon containers which may be handled mechanically or may be shot in and out of the reactor core by compressed gas. The irradiation tubes going into the core should be designed so that contamination of the containers is not a problem. The outlet of the irradiation tubes should feed either into a lead shield which may be stored until the irradiated isotopes are needed, or into a machine for remotely opening the containers and discharging the isotopes into the shielded container from which they will be used.

Large quantities of material to be irradiated are canned in aluminum or tin containers that are placed in tubes in the reactor core. On discharge, these containers are remotely ejected into the reactor canal. Under water, the sources are placed in suitable shielded containers for transportation.

In addition to the handling of irradiated sources from the nuclear reactor, the **spent-fuel elements** must be removed. The operational procedure to be followed in this instance is a problem of the design engineers, for the nuclear shield must be penetrated and the spent element removed remotely. If the entire core is to be changed simultaneously, a complete section of the shield will often be removed and the core removed remotely to a waiting shielded and cooled shipping container. Individual-fuel-element removal is often necessitated because of substandard reactivity or bursting, in which case the elements may be individually removed remotely and stored in the reactor canal. Air-contamination monitoring and radiation surveys should be carried out continuously whenever the shield is penetrated.

Experimental reactors involve problems which deserve special mention. These reactors are not provided with any means of cooling and, therefore, any significant power-generation rate will cause damage to the reactor from thermal stresses, with possible dissemination of radioactive materials. The cores of experimental reactors are subject to constant change, and the human errors associated with changing the core loading are an ever-present concern.

The fission-product build-up is not a great concern with experimental cores. However, there will be an appreciable gamma flux from the core immediately following shutdown. Further, high concentrations of **noble-gas fission products** and their daughters ($T_{1/2} = 30$ min) will be found in the air when unclad fuel is used, necessitating the use of respirators or exhaust ventilation upon first entry into the reactor vault (Bolton, P. R., *et al.*, *Aerosol Activity from Experimental Low Power Reactor Operation, Health Physics*, vol. 1, p. 135, 1958). The core containers become highly contaminated, and any material removed from the core should be monitored for both radiation and contamination.

The **reactor canal** is used as a source-handling and storage location, the water serving as a shield against radiation while giving working accessibility to the sources. Normally, a canal will be about 20 ft deep and will have a shallow end for handling weak sources. Because of the depth of water involved, cleanliness of the water is essential to ensure visibility. Circulating the canal water through glass-fiber filters,

as FG-25, and ion-exchange resins will tend not only to keep the canal water clean but also to aid in reducing the radioactivity concentration.

Major reactor accidents may occur if reasonable precautions are not taken. The temperature of the reactor core will rise sharply if a loss of coolant occurs. Even in a water-moderated and cooled reactor, which would go subcritical upon loss of coolant, the fission-product gamma heating (about 5 per cent of the total power) would cause the core to heat. In the extreme case, the core components would melt down and the volatile fission products would be released to the environment. The release would not be instantaneous; there would be time for evacuation of personnel. The spread of contamination may take place over a large area of the locality but will be acute only within a reasonable distance of the plant under the worst atmospheric conditions.

Either a reactor excursion or an increase in power beyond the capacity of the cooling system may result in damage to the core components from thermal shock and stress. The primary coolant system could rupture and the primary coolant be released. The combination of loss of coolant and serious damage would cause the core to go subcritical, but not until irreparable damage had been done to the reactor.

The trend in reactor design is to eliminate all possible means of escape of radioactive materials. Wherever possible, reactors and their associated equipment are "canned" in containers sufficiently large and strong to retain any volatile products if the core should melt down.

If a core should melt down and the **release of fission products to the atmosphere** should occur, by no means all the fission products would become air-borne. The iodine, bromine, xenon, and krypton fission products, being volatile, would be the primary escaping components; but some of the nonvolatile fission products and possibly some fuel would also escape by entrainment in these gases.

When one of the coolant channels in a reactor becomes deprived of coolant flow, hot channel results. This can occur either through uneven placing of the coolant channels or by a channel's becoming plugged after assembly. The fuel rods adjacent to the hot channel are deprived of the necessary cooling, causing them to heat to a dangerous point and rupture. The resultant fission-product release will contaminate the entire core and cooling system, greatly hampering any maintenance. The coolant lines should be monitored to detect any increase in activity of the coolant that might indicate fuel-rod ruptures.

The **selection of a reactor-plant site** should be made largely on the basis of the proposed use, power, and type of the reactor. If an air-cooled reactor should be located in an area of frequent temperature inversions, its operations may be restricted. A nonpower water-cooled reactor should be located on a river to provide a large dilution factor for cooling water. Furthermore, any use of the water made down the river must be taken into consideration. If the water is used for irrigation, public water supply, or recreation, procedures must be established to monitor the release of radioactive materials into the river. The location and size of the reactor-plant site are normally governed by AEC approval.

INDUSTRIAL RADIOGRAPHY

Industrial X-ray equipment is used in industrial plants primarily for purposes of inspection, quality control, and analysis. It provides an excellent means for the nondestructive testing of a variety of materials to detect cracks, holes, or inclusions. Welded joints are frequently inspected in this manner to ensure that the weld does not contain any significant flaws. Standards for interpretation of flaws are found in the ASME Boiler Code. X-ray diffraction is useful for a variety of analyses. See Sec. 13 for information on this use.

X-ray apparatus provides a source of penetrating radiation having a high intensity which can be turned on or off at will. This ability to "turn off" the radiation is a major difference between X-rays and gamma rays from radioactive materials such as radium or Cobalt-60. Available for industrial radiography is X-ray apparatus which produces beams with energy ranging from a few thousand volts (kilovolts) to several million volts (megavolts). The radiation intensity in the useful beam is often as high as several hundred roentgens per minute. Personnel who operate such apparatus and are considered to be radiation workers must not receive a radiation dose greater than permitted by the rules of NBS Handbook 59 (or 60). Because of the high intensity of the useful beam from such apparatus, it is obvious that the X-ray installation must be designed and operated in such a manner that it is impossible for personnel to get any part of the body in the useful beam. The scattered X-rays from these high-intensity beams are so great that shielding personnel from the scattered radiation becomes a major criterion in the design, operation, and installation of industrial X-ray apparatus. Since the protection of personnel against overexposure to radiation is of prime consideration in the use of X-rays for industrial radiography, the complete requirements for this protection must be factored into the original equipment selection, proposed method of use, and installation design. If this is not done, the original usefulness may not be fully realized because of limitations made necessary for personnel protection. The many complicated engineering and technical factors involved in designing an X-ray installation, establishing operating practices, and evaluating actual radiation exposure of personnel will dictate the use of a qualified expert for evaluating and establishing the required protection measures.

X-ray apparatus gives a radiation beam containing rays which have energies extending from a few kilovolts up to the maximum specified for the equipment. The kvp (kilovolt peak) is the maximum energy of the radiation in the beam, the average being much less. It is customary to use filters of aluminum or copper in the useful beam to filter out the very low energy radiation, since this component usually serves no practical purpose. Thus, the actual energy distribution in the useful beam from an X-ray tube will depend upon the amount of this filtration used. The filters raise the average or mean energy of the beam but do not affect the maximum.

NBS Handbook 60 is primarily of value in connection with medical X-ray installations. See Sec. 3 for pertinent sections of Handbook 60 and tables used in calculations of **X-ray shielding**. Parts I and IV of Handbook 60, however, contain useful information applicable to industrial radiography. The ASA Safety Code for the

Industrial Use of X-rays, Z54.1-1946, is a sound basis for providing safety in X-ray installations used for industrial radiography. This code is out of date, however, and is being revised at the time of publication of this handbook. When the revision is completed, this ASA Code Z54.1 should be used as the basis for providing protection of personnel. If the Z54.1-1956 code is used, the permissible daily dose should be changed to conform to the newer permissible radiation exposure limits established in NBS Handbook 59 (or 60).

In ASA Code Z54.1-1946, a classification of industrial X-ray installations is made as follows: class A, totally protective installations; class B, protective installations; and class C, all other installations. Specific requirements for the design and installation requirements of each class of protection are given, and restrictions on operation of each class are provided. There has been a tendency to require that all new installations be made to conform to class A requirements. Depending upon the nature of the work and the type of operating personnel, the class A installation may be required or may prove to be the most economical over a long period of time. In many instances, however, the restrictions imposed upon the operation of a class B installation may be unimportant for the particular work, and a class A installation would be both uneconomical and unjustified. Class C installations should be considered only when a qualified expert has carefully analyzed the type of work to be performed and specifically recommended its use.

X-ray equipment used for industrial radiography is usually operated part time. Since **radiation exposure of personnel** occurs only while the apparatus is operating, a higher *rate* of exposure may be permitted than would be the case for continuous operation; however, this would place it in the ASA class B category. For this reason, the "work load" factor is introduced into the protection requirements specified in NBS Handbook 60.

Exposure of personnel is specified in terms of quantity of radiation absorbed in the tissue during a given time interval—not the instantaneous rate.

Although it is desirable and frequently possible to place industrial X-ray equipment at one location and bring all the materials to be inspected to this one location, there are many instances where this is undesirable or impractical. Portable or movable X-ray units are then taken to the location of the work to be inspected. Under these conditions, the provision of shield walls, interlocks, and other protective features is not possible. Whenever industrial X-ray equipment is set up and operated outside of a regularly established and approved room, extreme care must be taken to compensate for the lack of protection which is built into the permanent installation. A qualified expert should always establish the permissible operating procedures for portable equipment. Wherever practical, the useful beams should be directed into the ground. When this is not practical, extreme care must be taken to assure that personnel cannot enter the area traversed by the main beam under conditions which represent the minimum mass of intercepting or shielding material. Because of the high-intensity radiation in the useful beams, effective protection against any person's accidentally entering the main beam must be provided.

Scattered radiation may present a potential exposure risk, and protection against overexposure must be provided by either shielding or use of distance. Wherever distance is used as a means of preventing overexposure, physical barriers of some type should be established and proper warning signs installed to inform people of

the significance of the barrier. Although "roping off an area" may seem like sufficient protection against persons entering a high-radiation area, it must be realized that it is a simple and common matter to violate such a barrier, and continual surveillance is necessary to ensure that personnel respect it. Fences of reasonable height are better protection although not so flexible in use.

In addition to its use as a research device, **X-ray diffraction equipment** is used for a variety of purposes in industry. At the time such equipment is installed, a careful survey should be made to detect any leakage of radiation from the shields which are provided. These units are designed to provide a small-diameter useful beam in specified directions or locations. The equipment must be arranged so that personnel will not be exposed to this high-intensity beam; sufficient shielding should be provided so that there can be no significant exposure in any other direction. Particular care must be taken to ensure that there are no narrow beams caused by cracks or small openings in shields. Since there is a likelihood that this equipment may be disassembled and reassembled, particular care must be taken to ensure that an adequate survey is made each time the shield around the tube is changed. Since the safety of the operators of such equipment depends to a considerable extent on their operational procedures, it is important that such persons be thoroughly indoctrinated and have a complete understanding of hazards that may be associated with such equipment.

GAMMA-RAY SOURCES

Gamma-ray sources are used in industry primarily for inspection and quality control, but they also find a variety of other uses in calibration of radiation detection or measuring instruments, calibration of films, production of ionization in air or liquids, and in research applications. They are also being used experimentally for a variety of uses such as irradiation of foods and plastics.

In all these uses the basic requirements for protection of personnel, both those performing the work and others in the area, are essentially the same. The radiation exposure of persons who are considered to be radiation workers, and have been so advised, must be limited to 5 r/year. Exposure of other personnel who are not radiation workers is limited to 0.5 r/year. See Sec. 3 (NBS Handbook 59) for more detailed information on permissible exposures.

The following table gives the pertinent characteristics of the gamma-ray sources which are most commonly used in industry:

Source	Energy of gamma, Mev	Half-life
Radium.....	0.18, 0.24-0.35, 0.42-2.4, 5	1,622 years
Cobalt-60.....	1.17, 1.33	5.3 years
Cesium-137....	0.661	33 years
Iridium-192...	0.87	74 days

Sealed sources of gamma rays are ordinarily made by hermetically sealing the radioactive material in a small capsule which prevents the physical escape of any of the radioactive material but does not significantly reduce the gamma radiation.

Several of the commonly used isotopes also emit alpha or beta radiation, but the capsule is usually designed with sufficient thickness to stop these rays. The capsule must never be opened, except by its manufacturer or by persons adequately trained and equipped to prevent radiation exposure. Special care must be taken not to break or open the capsule while in use, and periodic tests should be made to ensure its integrity. Special care and procedures are required for radium capsules.

Since **radium** is a naturally occurring radioactive element, it was used extensively as a source of gamma radiation prior to the advent of nuclear reactors. Through experience, much information has been collected relating to the proper methods of manufacturing and using radium capsules. Much information can be found in technical literature.

Radium-226 decays by alpha emission to Radon-222, a chemically inert radioactive gas. In addition to the alpha emission, the Radium-226 has an 0.18-Mev gamma associated with 5.7 per cent of the total alpha disintegrations. The daughters of Radon-222 are actually the main sources of the gamma radiation from a radium capsule. It is apparent, therefore, that the radium must be adequately and completely enclosed and sealed in the capsule so that all the decay products (which must pass through the radon gas daughter) are retained and will come into equilibrium with the parent, Radium-226. Data on gamma radiation from a radium source are based on the assumption that the daughters are in equilibrium with the parent. Since the gamma rays come from several daughters and the daughters give rays of more than one energy, a wide range of energy will be found in the gamma rays from radium.

Since the effectiveness of **shielding materials** is dependent upon the energy of the radiation, it is impractical to select one energy value for use in designing the protective shields that are to go around radium. The characteristic of the shield must be determined for the spectrum of energies which come from radium and, therefore, special tabulations and graphs have been developed for evaluation of shields used with radium (see Sec. 8).

Radium is extremely toxic. Extreme care must therefore be taken to ensure that Radium-226 is handled, processed, or used in such a way that it cannot gain access to the body by ingestion, breathing of dust, or contamination of wounds.

The hazards from **encapsulated radium (or Cobalt-60)** sources are usually considered to be those of **external radiation exposure**. Other conditions being equal, the dose is directly proportional to the amount of radium and to the time of exposure. That is, doubling the amount of radium or the time of exposure doubles the dose. However, the dose is inversely proportional to the square of the distance from the radium to the part of the body under consideration. Thus, doubling the distance reduces the original exposure by a factor of 4. For this reason it is often said that "distance is your best protection" when handling radium. In any particular case it is the proper balance among distance, thickness of shield, and time required for the manipulation which will reduce exposure to the lowest practicable limit.

Internal radiation hazards exist when radium capsules develop leaks that result in inhalation or ingestion of radium-bearing dusts or decay products from radon gas. For this reason, it is extremely important that radium capsules be properly designed, sealed, and tested.

Internal pressure in radium capsules, which is frequently responsible for leaking sources, may be due to two causes:

1. The inevitable accumulation of helium which is in direct proportion to the age of the source and to the quantity of radium. Under ordinary conditions, the helium pressure should not become hazardous for many years, and estimates of this risk should be obtained from the manufacturer of the source.

2. Failure to allow radium salt to dehydrate thoroughly before it is sealed in the capsule can be the cause of even more internal pressure than can the accumulation of helium. The ultimate pressure will be independent of the quantity of radium.

When a radium capsule having excess internal gas pressure is subjected to mechanical stress or to sudden extreme temperature changes, the danger of capsule rupture is greatly increased and may occur suddenly with an accompanying ejection of some of the radium. Mechanical damage to the capsule may release some of the radium. Any opening in the capsule, no matter how slight, will release radon gas which, in turn, will contaminate the area with its daughters.

Rupture of a radium capsule is serious because it may result in contamination of the area and thus create health hazards. The handler must therefore be continually aware of any use of radium capsules which might subject them to stress, excessive wear, corrosion, or mechanical damage. For example, radium capsules are frequently employed for calibration of film badges or pocket dosimeters. A convenient method of positioning the radium capsule for such calibration work is the use of an "air lift" which operates by passing an air stream through a tube which has an internal diameter only slightly greater than that of the radium capsule. The air stream can be used to move the capsule from the shielded container to the calibration position. When the air is released, the capsule will fall back into the shield. This provides a simple method of timing the exposures and of providing maximum shielding for the source. It is apparent that the capsule is subjected to wear or mechanical damage in this arrangement, and the air stream passing the capsule will promptly disperse radium dust in the event the capsule is ruptured. Such a capsule failure can produce serious contamination problems. The rupture of a capsule under these conditions can cause monetary losses amounting to many times the value of the radium, and cause a complete disruption of the facilities to permit satisfactory decontamination. It must therefore have adequate mechanical strength and wear resistance.

Iridium is sometimes used as a source of gamma rays for radiography. An output of about 11 r/min at 8 cm can be obtained from a unit weighing only 50 lb. It has sufficient lead shielding to make pneumatic control of the source unnecessary.

Iridium presents primarily an external radiation hazard. There is very little possibility of an internal radiation hazard from gases that might escape the encapsulated source.

Cobalt-60 is used in industry for nondestructive testing, counting room standards, calibrating instruments, and measuring underground formations. In medicine it is used primarily in teletherapy units. Sources containing 2,000 curies of cobalt can be made which will deliver a dose rate of 50 r/min in air at a distance of 80 cm. Cobalt, in contrast to radium, has the following advantages:

1. It does not have the gaseous decay products; therefore, gasproof sealing is not necessary.

2. It can be machined into a wide variety of sizes and shapes before irradiation and activation.

3. Its cost is moderate and its supply practically unlimited.

4. Its permissible concentration in air is much higher.

5. It may be reirradiated any number of times.

6. It is magnetic; this permits another method of remote handling.

Cesium-137 is a by-product of the Atomic Energy Program, and because large quantities are available, investigations are now under way to find ways to utilize cesium as a substitute for radium sources in medicine and industry. It will most likely be used as a source in teletherapy units. Cesium, like cobalt, emits practically monenergetic gamma rays. The isotope is usually produced as powdered Cs_2SO_4 and then is utilized in a sealed container. The radiation stability of Cesium-137 must be carefully evaluated, since certain cesium salts decompose and evolve oxygen. Therefore, the hazards are similar to those of radium, but less severe.

All types of gamma-ray sources should be tested for contamination or leakage upon receipt. Sources certified by the National Bureau of Standards are so tested at the time of certification. If there is some reason to believe that a source has been damaged, it should immediately be tested for leakage.

To **test for leakage**, a Geiger-Müller detector, scintillation counter, or an alpha survey instrument is required. Each source to be tested can first be wiped or "smeared" with a filter paper using long-handled tongs or other adequate means of radiation protection. It can then be placed close to or wrapped in an absorbent material such as cotton or filter paper and left for at least a day, preferably in a small sealed container. The absorbent material should then be checked with a suitable instrument. The presence of contamination indicates a leak. If the capsule contains radium and leakage is gross or has existed for some time, merely smearing the source or its container and testing the smear should indicate the presence of contamination. Proper radiation-protection measures must always be taken during such testing.

Sources that leak should be placed in sealed containers and sent to the manufacturer or a qualified laboratory for repair and measurement. Containers and carriers, as well as any other equipment that has had contact with the leaking source, should be decontaminated under the direction of a radiation safety officer.

When a source container or seal has very small fissures, radioactive material sometimes leaks out quite slowly. It is therefore advisable to make two leakage tests at least 1 week apart before concluding that the source does not leak. Leakage testing for the majority of sealed sources is recommended at least once every 6 months.

If there is reason to suspect the leakage of a potentially hazardous amount of radioactive material may have occurred, the following emergency measures should be taken at once:

1. No immediate attempt should be made to clean up the spill.
2. All windows should be closed, fans and air conditioners should be shut off, and everyone should leave the room.
3. All doors should be closed and locked.
4. If powdered sources are involved, the door and all other openings leading into

the room should be sealed with wide masking tape or adhesive tape and heavy wrapping paper.

5. Every person who might have been in the immediate area must be considered contaminated until checked by a radiation safety officer.

6. Entrance to the contaminated area should be prohibited until a consultant experienced in radiation hazards can be called in and his advice followed.

Under no circumstances should any untrained person attempt to examine or clean up any **spilled radioactive material**. The cleanup technique should be planned with the same care used in quantitative chemical analysis or in bacteriological work with virulent organisms. Fans or ventilating apparatus should not be turned on in an attempt to blow away the radioisotopes, except possibly in special cases involving gaseous sources. Such a maneuver will only disseminate the radioactive materials through the area. If the radioisotope is blown out of a building, air currents may carry the finely divided material into nearby windows or air-intake ducts. Proper precautions taken immediately will protect human life and minimize financial losses.

Extreme caution must be exercised in the **handling of radioactive sources**. They should never be touched with the hands. There should always be some distance between sources and the operator. In the case of radium or other powdered sources, methods of transfer that subject the source to repeated shocks or vibrations should be avoided. Also, the transfer methods should provide automatic safeguards against damage to the source and against spread of contamination caused by malfunction of the systems.

Easily damaged sources should be lifted with forceps having a "spring tip" adjusted to prevent excessive pressures. "Cross-action" forceps are desirable for some delicate manipulations.

The handling normally done by forceps can often be more conveniently performed by vacuum-operated "hand guns" with pistol-type grip and trigger. One must be certain, however, that the gun is so designed that damage to the sources cannot occur.

If the sources are magnetic, magnetic handling devices may be used. Either the radioactive material itself may be magnetic, as in the case of Cobalt-60, or the container of the source may be magnetic. The residual magnetism of these sources should not be much higher than that of soft iron, or else it may become difficult to perform the necessary operations rapidly and accurately. If the sources become permanently magnetized to such a degree that they cling together or to any magnetic material with which they come in contact, their handling, cleaning, and storage will become difficult.

Storage of radioactive sources should be in containers, vaults, or rooms constructed in such a manner as to ensure that the dose rate does not exceed the appropriate limits as specified in NBS Handbook 59 (see Sec. 3).

Servomechanisms and rotating storage containers may be used to advantage in the selection and distribution of individual sources. Each container of radioactive material in storage should, in addition to the standard radiation-hazard symbol, be labeled in such a manner that the kind and quantity of material, the date of measurement, and the name of the person responsible for the material can be easily and quickly determined. Where a large number of sources are stored, a lead-lined safe with lead-filled trays may be used to advantage. This permits the individual

sources to be stored in holes in the lead of the trays. Separate compartments should be provided for different types of sources. Each compartment should be marked so as to permit immediate and certain identification of its contents from the outside. It is highly desirable that tubes, cells, needles, etc., be readily identifiable as to their type and activity from a considerable distance.

Shipments of radioactive sources should be made in accordance with the current Interstate Commerce Commission or Official Air Transport Regulations (see Sec. 3).

Scattered radiation must be considered in all work with gamma rays. If "shadow shielding" is employed, it may be easy enough to prevent exposure of personnel in the unshielded areas by means of barriers, but radiation reflected from the air or surrounding objects may create a hazard in the shadow itself. Any use of shadow shielding, therefore, must take into consideration the intensity of scattered radiation from the unshielded spaces.

SEALED BETA-RAY SOURCES

Encapsulated radioactive material is referred to as a sealed source. When the primary emission from the material is beta radiation the source is referred to as a **sealed beta-ray source**.

Pure beta emitters are desirable and are usually obtained by separation of mixed fission products. Strontium-90, Yttrium-91, Technicium-99, and Promethium-147 are the most common beta sources. Others are listed in *Nucleonics* (vol. 13, no. 6, p. 78, June, 1955).

One of the widest **uses of beta-ray sources** is the removal of static electrical charges from rapidly moving insulating materials such as paper or cloth passing through high-speed machines. The beta-ray source near the moving web ionizes the surrounding air and permits the static charge to be removed.

Beta rays have found some use as sources of energy for the "triggering" of chemical reactions. In this application, they are most useful in gas-phase reactions where the streams of gas can be passed around the beta-ray source. They can also be used in liquid-phase reactions if the liquid layer is kept very thin.

A small beta emitter in a gaseous-discharge tube permits faster start-up time and a lower starting voltage because it ionizes the gas in the tube before the starting potential is applied.

Other uses of beta-ray sources include gauging of liquid levels in tanks, gauging thicknesses in sheet metals or plastics, production of self-luminous markers, and production of nuclear batteries.

The chemical and physical form of the radioisotopes contained in sealed beta-ray sources should be chosen to:

1. Minimize the occurrence of chemical or radiation damage which may cause the sealed source to leak radioactive material. Nitrates and organic compounds are usually unacceptable from this standpoint.
2. Minimize the likelihood of intake into and retention in the body.
3. Minimize dispersion of the radioactive material if the container should leak or be damaged by accident or fire.

Solid radioactive metal, foil, or electroplated sources are the most desirable **forms of beta-ray sources**. Other forms include fused glass or ceramics with the radio-

isotope as one constituent, radioactive powder bonded to a ceramic base, and radioactive powder covered with a thin gold ribbon. If practical, source materials in the form of gas, liquid, readily decomposed solids, or loose powder should be avoided. In cases where more than one radioisotope will satisfy the functional requirements of a particular application, preference should be given to one with the lowest toxicity.

Although a sealed source is designed to prevent leakage of the radioactive material, the encapsulating material on a beta-ray source must be quite thin and, as a consequence, can be easily damaged. It should be designed to minimize the probability of leakage of the radioactive material in normal use, shipment, and under such abnormal conditions as can be foreseen. Disassembly of the source should be made virtually impossible by using construction methods such as welding, riveting, crimping, soldering, and peening. Also, careful consideration should be given to possible eventual damage to the source or its seal because of any of the following factors:

1. Radiation from the source itself, either direct or indirect, which accelerates corrosion either directly or by production of corrosive ozone in the air
2. Attack by chemicals inside the source
3. Attack by corrosive fumes, solvents, or other chemicals to which the source may be exposed
4. Build-up of gaseous pressure inside the encapsulating material by the action of radiation or by heating, such as in a fire
5. Breakdown due to discharge of high electrical potentials built up by the transmission of beta particles through insulating material
6. Vibration, shock, or other mechanical injury
7. Stresses set up by differences in thermal expansion
8. Deterioration inherent in the materials used for the container (e.g., loss of solvents or plasticizers from plastics)
9. Damage from high or low temperatures, humidity, low pressure experienced in shipment by air, or any other unfavorable environmental conditions which might be foreseen

Because many of the design factors are difficult to evaluate thoroughly, it is strongly recommended that tests be made by the manufacturer on a prototype of each new sealed source before it is put into industrial use.

When a sealed-source window or seal has very small fissures, radioactive material sometimes leaks out quite slowly. It is therefore advisable to make two **leakage tests** 1 week apart before concluding that it does not leak. For the majority of sealed sources, leakage testing by the user at least every 6 months is recommended. Special leakage tests should be made after accidents, shipments to a new location, and at such other times when environmental conditions might cause damage to the source seal.

If there is reason to suspect that leakage of a potentially hazardous amount of radioactive material may have occurred, the following **emergency measures** should be taken at once:

1. No immediate attempt should be made to clean up the spill.
2. All windows should be closed, fans and air conditioners should be shut off, and everyone should leave the room.

3. All doors should be closed and locked.
4. If powdered sources are involved, the door and all other openings leading into the room should be sealed with wide masking tape or adhesive tape and heavy wrapping paper.
5. Every person who might have been in the immediate area should be considered contaminated until checked by a radiation safety officer. They should stay in the immediate vicinity to prevent spreading the contamination.
6. Entrance to the contaminated area should be prohibited until a consultant experienced in radiation hazards can be called in and his advice followed.

Under no circumstances should any untrained person attempt to examine or clean up any spilled radioactive material. The cleanup technique should be planned with the same care used in quantitative chemical analysis or in bacteriological handling of virulent organisms. Fans or ventilating apparatus should not be turned on in an attempt to blow away the radioisotope, except possibly in special cases involving gaseous sources. Such a maneuver will only disseminate the radioactive materials throughout the area. If the radioisotope is blown out of the building, air currents may carry the finely divided material into nearby windows or air-intake ducts. Proper precautions taken immediately will protect human life and minimize financial losses. Any sealed source found to have a leak should be removed from use and placed in a sealed cask pending return to the supplier or disposal.

Suppliers of sources usually attach a durable label to each sealed source or sealed-source container indicating the manufacturer's name, the serial number of the sealed source, the chemical symbol, atomic weight of the radioisotope contained within, the curie strength of the source, and tolerance distance.

Caution signs bearing the conventional magenta radiation insignia should be conspicuously posted near the operating equipment using a beta-ray source, with an indication of the minimum distance which should be maintained between employees and source locations. It is desirable to have any hazardous area defined with barriers as a physical reminder to personnel. The names of personnel to be notified in case of a radiation emergency should be included on or near the caution sign.

Storage of a sealed source not in active use should be such as to prevent tampering or handling by unauthorized persons and to keep personnel exposure to an absolute minimum and never in excess of the maximum permissible weekly dose. The container should be prominently marked and secured in such a manner that only authorized persons have access to the source.

Shipping of radioactive materials must be in strict accordance with the latest pertinent regulations of the Interstate Commerce Commission, the Post Office Department, or the Civil Aeronautics Administration, whichever is applicable (see Sec. 3).

Prior to shipping, the sealed source should be so packaged as to ensure safety for both the transportation company and the receiver. Each sealed-source container should be free of loose surface contamination and fixed radioactive material.

Sources which require disposal because of leakage, damage, or having served their useful life should preferably be returned to the manufacturer. Where this is impossible, they may be disposed of in a manner approved by the appropriate governmental agency.

Before a person is allowed to handle radioactive sources, he should be informed of the hazards involved and how to guard against them.

Personnel monitoring should be employed where radiation safety depends upon proper operating procedures. Film badges may be used and pocket dosimeters will be of value if the beta energy is high enough (approximately 1.5 Mev or higher). Various types of wrist badges or rings incorporating films are available for monitoring local exposure to the hands. If a particular operation is routine or repetitive and the hazard is slight, the exposure may be established without the necessity of wearing such devices continually.

In every plant using radioactive sources which constitute a potential hazard to personnel, the responsibilities for radiation protection should be clearly defined by management and some responsible employee should be designated as radiation protection officer. The radiation protection officer should be made responsible for the establishment of satisfactory working conditions according to the NBS Handbooks (see Sec. 3).

Special physical examinations, other than those considered to be good medical or industrial medical practices, should be necessary for persons using sealed beta-ray sources. Preplacement physical examinations are always advisable in order to reveal any physical conditions that might later be attributed to radiation exposure or that might otherwise make it undesirable for an individual to be exposed to radiation. The preplacement examination should include a general physical examination, medical history, radiation exposure history, and a blood count. If there is any possibility of accidental overexposure of any person, a preplacement blood count for any such individual might be useful as a later reference to indicate the presence of significant radiation damage.

If "bone seekers," such as Strontium-90, are inhaled, ingested, or enter the body through a wound, prompt medical attention can markedly reduce subsequent deposition in the bones. The physician can obtain up-to-date information on such treatment promptly by telephone from the Director, Division of Biology and Medicine, U.S. Atomic Energy Commission, Washington, D.C., or from the medical director of one of the AEC subcontractors.

RADIOACTIVE ISOTOPES

Nearly 1,000 **radioactive isotopes** have been derived from the elements found in nature. They are produced by bombarding materials with high-speed subatomic particles in devices such as the cyclotron, by subjecting materials to the bombardment of the neutrons emitted by fission in reactors, and by separation of the fission products from spent reactor fuels. Technically, all radioactive materials are radioactive isotopes and the subject "radioactive isotopes" might be construed to encompass all use of radioactive materials. This name, however, has come to mean a limited number of these materials which have been prepared in specific forms for a variety of experimental or technical uses and which are usually handled in amounts of less than a few curies. The natural radioactive materials are usually not included. Some are prepared for specific applications such as gamma-ray sources or beta-ray sources. Many of these "radioisotopes" are

used in industrial or research laboratories, and their use in industrial processes is increasing.

Where radioisotopes are used in laboratories or plants that have a competent industrial-hygiene or health-physics department, the control of health hazards from them will usually be integrated into the policies and procedures that are used throughout the facility. Many radioisotopes, however, find uses in locations where existing personnel are unfamiliar with the many ramifications involved in the control of such hazards. It is highly desirable to have any such proposed use of radioisotopes reviewed by competent industrial-hygiene or health-physics personnel. Such assistance can be obtained from insurance companies, universities, state industrial-hygiene departments, AEC Isotopes Division, or consultants. The judgment of such a competent person will ensure that adequate health-protection measures are incorporated and, at the same time, prevent unnecessary expense from overconservative control measures. The following general information cannot be considered as complete data on all phases of isotope use but will serve to outline some general information that will be useful to radioisotope users.

The one property of radioisotopes which makes them extremely useful tools for research and industry—that is, the property of radiation—is the same thing which makes them potentially hazardous to the health of the persons using them or coming in contact with them. Although a large number of radioisotopes are available, there are only three types of radiation emitted by them, namely, alpha, beta, and gamma.

The hazards in **handling radioisotopes** may be classified in the order of their importance, as follows:

1. Internal exposure from deposition of radioisotopes in the body by breathing, ingesting, or absorbing them
2. External exposure of the whole body or limited parts of the body to beta or gamma radiation from the radioisotopes
3. Radioactive surface contamination of the body, facilities, and equipment

Some radioisotopes are more hazardous than others. The **relative hazard** of some of the beta-gamma isotopes is shown in Table 2 of NBS Handbook 42, "The Safe Handling of Radioactive Isotopes" (see Sec. 3). The least hazardous can be safely handled or used in small quantities with very limited precautions. The most hazardous ones, in larger quantities, require very elaborate precautions and highly specialized equipment. The alpha-emitting isotopes also vary considerably in hazard. Unless they also emit significant beta or gamma radiation, they present only an internal radiation hazard. Data on maximum permissible concentration in air and water can be obtained from Table 3, NBS Handbook 52 (see Sec. 3). For occupational exposures the values in Table 3 should be multiplied by 3, as indicated in Appendix 3.

Factors determining the toxicity of an isotope which enters the body include its physical half-life, biological half-life or rate of elimination, energy and type of radiation, the site of deposition or critical organ in the body, and the amount which is deposited in the critical organ. The most hazardous isotopes are those which have a long physical and biological half-life with respect to life span, and a high rate of deposition in small organs, radiosensitive tissue, and bone.

In **chemical laboratories**, the main route of entry of radioactive materials into

the body is by inhalation of radioactive air-borne dusts, mists, fumes, vapors, and gases. The ingestion of radioactive isotopes is usually of secondary importance as a route of entry to the body. In routine laboratory operations, the amount ingested is normally quite small if reasonable handling precautions are exercised. However, considerable care must be exercised to avoid the accidental ingestion of considerable quantities such as could occur when pipetting radioactive solutions by mouth. Small amounts of radioactive material may be ingested by the transfer of contamination from the hands and other objects brought to the mouth, or as a secondary result of inhalation. Some radioactive materials can enter by absorption through an open cut or even through the intact skin.

Isotopes which are beta or gamma emitters may present an **external radiation hazard** if the quantity handled and the energy of the radiation are sufficiently great. Film badges should be used for measuring beta or gamma exposure to ensure that the maximum-permissible-exposure levels are not exceeded. Pocket dosimeters are very useful but will not record low-energy beta radiation. Survey meters can be used to advantage in establishing appropriate levels of radiation intensity wherever beta-gamma radiation exposures might be significant. External radiation exposure should be controlled within the limits specified in NBS Handbook 59, "Permissible Dose from External Sources of Ionizing Radiation" (see Sec. 3). Alpha-emitting materials, if kept outside the body, are not hazardous because of the short range and low penetrating power of alpha particles. Film badges or pocket dosimeters do not record alpha radiation, and where only alpha-emitting materials are used, they are of no value.

Contamination is defined as the presence of unwanted, unconfined, and uncontrolled radioactive materials. Contamination on a person or his clothing can enter the body by transfer to the mouth, by breathing dust from it, by absorption from a cut, or for a few compounds, by absorption through the skin. Beta- and gamma-emitting isotopes on a person can contribute to his external radiation exposure, and retention of radioactive materials in the skin itself may produce local radiation damage. Contamination on laboratory facilities and equipment can easily be transferred by physical contact or by room air currents to the person, to offices, and to other equipment and laboratories where the presence of contamination would interfere with analytical results or cause damage to some manufactured product such as film. Extensive contamination of large areas of the laboratory, such as walls, floor, and benches, can contribute to the external radiation exposure, since persons working in the area are essentially surrounded by a source of radiation.

The control of hazards requires that operations be done in a laboratory which is properly equipped for handling radioisotopes (see Sec. 3). The type of equipment to be employed depends upon the relative hazard of the isotope, the physical form, the quantity to be handled, and the operations involved. For example, a powdered Strontium-90 compound requires more elaborate handling facilities than a Sodium-24 compound in solution. The alpha-emitting isotopes and the more hazardous beta or gamma emitters may require that all operations be performed in a **dry box** which is maintained under a slight negative pressure (Kelman, Wilkinson, Shuck, and Goertz; Handling Alpha-active Pyrophoric Materials, *Nucleonics*, vol. 14, pt. 1, no. 3, p. 61; pt. 2, no. 4, p. 65; pt. 3, no. 5, p. 77; March, April, May, 1956). Shielded dry boxes may be required for large quantities of gamma emitters. In some cases,

it may be necessary to perform the operations on the gamma emitters by remote-control devices built into the dry boxes to prevent excessive external radiation exposure to the body or limited parts of the body. External radiation can be controlled by providing adequate distance from the radiation source, limiting the time of exposure to the radiation, reducing the amount of radiation by shielding, or by combinations of these three.

Where dry boxes are not required, **laboratory hoods** similar in design to the usual chemical hood will be useful if dust or fumes are produced. Special hoods for work with radioisotopes are available and incorporate many useful features such as ease in decontamination and proper air flow. **Filters** may be needed on these hoods if the air discharge from them would contain sufficient radioactive material to exceed the permissible level for discharge (see Sec. 22). The filters should be easily accessible, and provision should be made so that removal of the filters will not cause contamination of laboratory areas, work areas, or personnel. Possible advantages of mounting filters in a separate room or in a location where they can be easily removed without causing contamination of laboratory or work facilities should be considered.

Operations which produce air-borne contaminants such as dusts and mists should be avoided where possible. Handling the isotopes in wet form or as solutions practically eliminates the dust problem. The selection of equipment and operations which have the least tendency to produce air-borne contamination aids in alleviating the problem. When these methods of control are not adequate, ventilation of the aerosol-producing operation is required. Local exhaust ventilation is preferred to general room ventilation since the former method collects the contaminant at the source, providing more reliable control of the hazard and minimizing the spread of contamination.

Even with the best facilities and equipment, the radiation hazard cannot be effectively controlled without the establishment of good work habits and good house-keeping practices.

Good **work habits** involve (1) a thorough understanding of the job and its attendant hazards and (2) a strict adherence to established safe procedures. In cases where the consequences of an accident would be severe, it is advisable to make a "dummy" run of the procedure in detail, using all the equipment and materials except the radioisotope. Such a trial affords an opportunity to correct any deficiencies in the procedures or equipment before introducing a radiological hazard.

Good **housekeeping** is essential for the control of radioactive contamination. This entails keeping the work area neat and orderly and free of extraneous material and equipment. Spills of radioactive isotopes are to be avoided, but when they do occur, they should be confined to an area as small as possible and should be cleaned up as soon as possible. Adequate precautions should be taken to prevent excessive radiation exposures during any cleanup operations.

Spills can be avoided by transporting the radioisotopes in suitable containers which will not break or come open if dropped. The use of trays under equipment to catch spills is recommended. Covering the working surface of laboratory benches, hoods, etc., with an absorbent paper prevents contamination of these surfaces, absorbs liquid spills, and helps retain spills of dry materials.

To confine the radioactive material to a specific area, it is necessary to check and

decontaminate, if necessary, all equipment and materials leaving the area. Contaminated materials and equipment should not be released to channels of commerce, the public, or other nonradioactive areas in the laboratory (see Sec. 19).

The use of **protective clothing** is frequently required to prevent contamination of the body and personal clothing. Such clothing should be removed at the exit of the contaminated area and be deposited in suitable containers. This practice will aid in preventing the spread of surface contamination. The amount of protective clothing required will depend upon the quantity, form, and toxicity of the isotope used; the handling facilities; and the type of operations to be performed. As a minimum precaution, a laboratory gown should be worn over the personal clothing. At the other extreme, complete protective clothing including shoes or shoe covers may be needed (see Sec. 18).

In a chemical laboratory handling radioactive isotopes, **waste disposal** is not so great a problem as in manufacturing operations, since smaller quantities of material are processed and the half-life of the isotopes is frequently relatively short. Wastes which are contaminated with short-half-life isotopes can usually be retained until the activity has decayed to a level which permits disposal by conventional means. Wastes contaminated with long-half-life materials or large quantities of shorter-half-life isotopes can be disposed of by commercial radioactive-waste-disposal services. Further information can be obtained through commercial channels or from Isotopes Extension, Division of Civilian Application, Oak Ridge, Tenn. It is generally advisable to minimize the volume of wastes requiring special treatment or disposal. In some cases, the waste-disposal problem may be alleviated or eliminated entirely by taking reasonable advantage of the environmental factors of dilution, dispersion, etc. The method of disposal selected usually is dictated by the quantity and hazards of the isotope used and the form, quantity, and contamination level of the wastes produced (see Sec. 21).

Labeling of radioactive isotopes is necessary to prevent their misuse.

Phosphors activated with radium have been used for many years on instrument dials, markers, signs, and for similar purposes. More recently, pure beta-emitting fission products have been used for such purposes because of their lower cost. These fission products must be of high purity since many foreign materials poison the phosphors. Since beta particles are less efficient than alpha particles for activating purposes, larger quantities of the beta emitters are required to produce the same amount of light which is obtained from a smaller quantity of an alpha emitter. Strontium-90, Carbon-14, Calcium-45, tritium, Thallium-204, and Promethium-147 have been used as the activators in standard light sources, self-luminous markers, paints, etc.

The hazards associated with the **manufacture of self-luminous items** and materials are dependent upon the activating radioisotope employed. In general, the most significant hazard is that of deposition of the radioisotopes in the body. This is particularly true for isotopes such as radium, Strontium-90, Calcium-45, and Carbon-14. Tipping of brushes in the mouth during **radium dial painting** operations produced many serious cases of poisoning (Martland, H. S., *Occupational Poisoning in Manufacture of Luminous Dial Watches*, *J. Am. Med. Assoc.*, vol. 92, pp. 466-473, 552-559, 1929). In the case of radium, there is an additional potential **inhalation** and contamination hazard from the radon daughter which may require

special ventilation controls. There is a potential external radiation hazard from the gamma-emitting radium and its daughter products. The external radiation hazard from the pure beta-emitting isotopes is of a much lower magnitude.

Self-luminous items such as light sources, markers, and dials are generally hermetically sealed by a plastic covering or are in other transparent containers of sufficient thickness that the beta radiation from them is insignificant. The gamma rays from such items employing radium as an activator would not be measurably reduced by the shielding of the container, but the quantity of radiation from a single unit is usually of no consequence. However, when large numbers of units are stored or used in one location, the radiation field may be of sufficient intensity to warrant the control of exposures of persons in the vicinity.

The **repair of self-luminous items** singly would present no great health hazard. However, persons engaged in the routine repair of such items where the luminous material is involved in the work should take adequate precautions to prevent the ingestion or inhalation of the luminous material and the contamination of the work area. Again, the storage of a large number of such items in the work area may result in some exposure to external radiation.

The **disposal of self-luminous items** should be in accordance with established practices for the particular radioisotope involved. Usually, the amount of radioactive material in dials, markers, and similar items is so small that no special precautions are required for disposal of a few items. The ICC Regulations exempt manufactured articles such as dials from specification packaging or labeling unless the gamma radiation at the surface of the package is greater than 10 mr per 24 hr.

THORIUM

Thorium has been used in industry for many years in limited quantities. It has been used in making gas mantles, electronic tubes, and more recently for **thoriated tungsten welding electrodes**. Although it is not fissionable, neutron irradiation of thorium will produce Uranium-233, which is fissionable, and the Uranium-233 can be used as a reactor fuel. This has resulted in increased use of the material.

Like uranium, thorium is a radioactive element occurring in nature. It has a long list of radioactive daughters, one of which is a noble gas, thoron. Since to this time no significant mineral deposits of thorium have been found in the United States, essentially no mining operations are involved here. There are small amounts of thorium in soil and ground water. Thoron and its decay products are found in small and variable amounts in outdoor air (see Secs. 4 and 6).

The potential health hazards when used in industry may be related either to the element, with all its decay products, or only to thoron and its daughters. Since the metal or its compounds continually emit thoron gas, there may be a hazard from breathing this gas and its decay products even though there is no dust or other airborne contamination. Even in freshly separated thorium, an equilibrium concentration of thoron builds up in a few days. Thoron daughters reach equilibrium values in a short time.

The rate of release of thoron from a sheet of thorium metal has been determined by Shlaer (Shlaer, Simon, personal communication, Health Division, Los Alamos Scientific Laboratory) to be 2.8×10^{-12} curie/sq cm of exposed metal under equi-

librium conditions. This test was run on metal which was of such age that it would produce concentrations one-half the maximum to be expected from fresh metal or very old metal. When enough sheets of thorium metal were suspended in a sealed room to produce a thoron concentration of 10^{-11} curie/liter of air, only about 6 per cent of the thorium B was found to be air-borne under stagnant-air conditions, the rest presumably having deposited out on the walls, metal, and floor. When a fan was introduced to stir the air, the value fell below 1 per cent. Under these test conditions, there was a relatively high surface-to-volume ratio in the room. When thoron was introduced by boiling a thorium nitrate solution in this same room without the presence of the metal sheet, about 30 per cent of the Thorium B was found to be air-borne in stagnant air. Under these conditions, the surface-to-volume ratio was much lower.

After its chemical separation, the specific alpha and beta activity in thorium varies considerably with time. Before separation, there are a total of six alpha and four beta disintegrations in the mixture for each Thorium-232 disintegration. After chemical separation, the ratios change with time as shown in Fig. 11-4 (Albert *et al.*, *AMA Arch. Ind. Health*, vol. 11, p. 235, 1955). Even after chemical separation, there are always two alpha disintegrations for each alpha which comes from Thorium-232. The second alpha comes from the short-half-life Thorium-228 which is present. When equilibrium conditions exist, the specific activity of natural thorium is 0.50 alpha disintegrations/min μg .

As with many other radioisotopes, the maximum allowable concentration for thorium in air has been subject to some question and numerous estimates. Some attempts to determine the maximum allowable concentration by calculation have indicated that the value might be quite low. A reliable industrial-hygiene and medical survey of a **thorium refinery** has been reported by Albert (Albert *et al.*, *Industrial Hygiene and Medical Survey of a Thorium Refinery*, *AMA Arch. Ind. Health*, vol. 2, p. 234, March, 1955). It was found that operations had been carried on for 30 years with exposures considerably in excess of currently acceptable standards for uranium. However, there was no evidence of any overt industrial disease. In their analysis of information relating to maximum allowable concentration for thorium, the authors indicated that it would be advisable to keep the permissible level of thorium dust equal to that of uranium on an activity basis. The maximum allowable concentration listed in NBS Handbook 61 is 200 disintegrations/min/ m^3 , and this seems to be a good estimate for thorium based on available information.

Thoriated tungsten electrodes have gained widespread acceptance since their introduction several years ago for use in inert-gas arc welding. The electrodes usually contain from 1 to 2 per cent of thoria, and the consumption of the electrodes

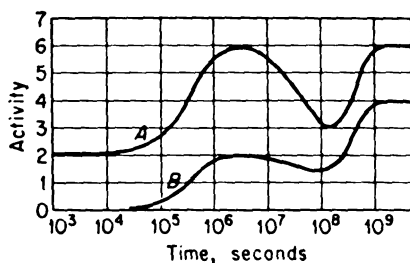


FIG. 11-4. Numbers of alpha (A) and beta (B) particles released from an equilibrium mixture of Th-232 and Th-228 for each decay of Th-232 at the indicated times after initial separation. (*AMA Arch. Ind. Health*, 1955, p. 235, Fig. 2.)

is very small. Breslin and Harris (Use of Thoriated Tungsten Electrodes in Inert Gas Shielded Arc Welding—Investigation of Potential Hazard; *Am. Ind. Hyg. Assoc. Quart.*, vol. 13, no. 4, pp. 191-195, December, 1952) reported on an investigation into the use of thoriated tungsten electrodes, and from their conclusions, it was evident that inert-gas welding with thoriated tungsten electrodes may be employed with no significant hazard to the operator or to other room occupants. Special ventilation or protective equipment is not indicated for protection from radioactive fumes. Similar observations and conclusions have been reached from other studies not recorded in the literature.

The concentration of thorium dust or fumes in the air can be determined by collecting samples from air on filter paper or plastic membranes and counting the samples for alpha activity. If any significant amount of the short-half-life daughters of thoron is present, they would be indicated by first counting the sample within a few hours of collection and subsequently counting it 1 or 2 days later after the thoron daughters have decayed.

General References

- Patty, "Industrial Hygiene and Toxicology," Interscience.
Yaffe, Byers, and Hosey, "Encyclopedia of Instrumentation for Industrial Hygiene," University of Michigan.
Drinker and Hatch, "Industrial Dust," McGraw-Hill.
Hemeon, "Plant and Process Ventilation," The Industrial Press.
Brandt, "Industrial Health Engineering," Wiley.
National Committee on Radiation Protection Handbooks, published as NBS Handbooks (see General Index).
Trasko, "Occupational Health and Safety Legislation," U.S. Public Health Service. ASA Codes (see General Index).
Mallette, "Air Pollution," Reinhold.
Interstate Commerce Commission Regulations, H. A. Campbell's Tariff 10.
"Handbook of Federal Regulations Applying to Transportation of Radioactive Materials," U.S. Atomic Energy Commission.
"U.S. Postal Guide," pt. I, art. 37, chap. 4, 1951.
"Radiological Health Handbook," U.S. Department of Health, Education and Welfare, Public Health Service, PB121784, January, 1957.
Lapp and Andrews, "Nuclear Radiation Physics," Prentice-Hall. Tannenbaum, "Toxicology of Uranium," National Nuclear Energy Series, McGraw-Hill.
Voegtlin and Hodge, "Pharmacology and Toxicology of Uranium Compounds," National Nuclear Energy Series, McGraw-Hill.
Hine and Brownell, "Radiation Dosimetry," Academic Press.
Murray, "Nuclear Reactor Physics," Prentice-Hall.
Lawrence and Hamilton, "Biological and Medical Physics," Academic Press.
"Industrial Utilization of Radioisotopes 1946-1956," A Forum Survey, Atomic Industrial Forum, Inc., August, 1956.
Guest, G. H., "Radioisotopes—Industrial Applications," Pitman, 1951.
Bradford, J. R., *et al.*, "Radioisotopes in Industry," Reinhold, 1953.
Morgan, K. Z., *et al.*, "Health Physics Insurance Seminar," TID-388, Office of Technical Services, Department of Commerce, Washington, D.C.

Section 12

RESEARCH APPLICATIONS

By

FREDERICK P. COWAN, Ph.D., *Head of Health Physics Division, Brookhaven
National Laboratory.*

Radiation Safety.....	12-2
Special Problems of Research Laboratories.....	12-3
Techniques of Radiation Protection.....	12-5
Safety Problems of a Research Reactor.....	12-16
Accelerator Safety.....	12-20

RESEARCH APPLICATIONS

Frederick P. Cowan

RADIATION SAFETY

See page 12-24 for References.

The safety of workers in respect to hazards such as fires, toxic fumes, and personal injury has long been a concern of both industrial firms and research laboratories. However, with the advent of greatly expanded work with radiation and radioactive materials since World War II, greatly increased emphasis has been placed on radiation safety. The term **health physics** has been widely applied to the broad field of diverse activities concerned with this problem. Health physics has been defined by H. M. Parker as the "borderland between industrial medicine, radiobiology, industrial safety, public health, physics, engineering and chemistry," and one might well add meteorology, geology, ecology, pedagogy, and applied psychology. Having connections with so many fields of science and engineering, it is natural that various aspects should enlist the interests of many specialists. Depending on the circumstances of organization, these specialists may be known as health physicists or by the designations of their specialties. Thus, considerable variation in emphasis and the details of organization is found at various laboratories.

The need for special emphasis on radiation safety arises from two important facts that should be fully understood by all radiation workers. In the first place, exposures far above the maximum permissible level produce no immediate physiological response to warn the individual concerned, although very harmful results may appear years later. Thus, safety can be achieved only by the use of instruments to evaluate the hazards involved. In the second place, the **toxicity of radioactive substances** is very high. This second point deserves to be emphasized here and as a part of the indoctrination of all workers with radioactive substances. Table 12-1 shows a selection of maximum permissible concentrations of radioactive substances in air, expressed in parts per million by weight as well as in terms of radioactivity, namely, microcuries per cubic centimeter. When we consider that the maximum permissible concentrations of even the more toxic nonradioactive chemicals are of the order of 0.1 ppm, we can appreciate the difficulty of working safely with appreciable quantities of radioactive materials. In this connection, it is instructive to compute the quantity of Strontium-90 that will contaminate the air in a medium-sized laboratory (20 by 30 by 10 ft) to the extent of 6×10^{-10} microcurie/cc, which is the maximum permissible concentration level for 40 hr/week occupational exposure, i.e., three times the value given in NBS Handbook 52. This amount turns

Table 12-1. Toxicity of Selected Radioactive Substances

Isotope	Max permissible concentration in air, microcuries/cc *	$T_{1/2}$	G/curie †	Max permissible concentration in air, ppm
H ³	2×10^{-5}	12.4 y	1.04×10^{-4}	1.73×10^{-6}
Ci ¹⁴	10^{-6}	5,600 y	0.22	1.83×10^{-4}
Na ²⁴	2×10^{-6}	15 h	1.15×10^{-7}	1.92×10^{-10}
P ³²	10^{-7}	14.3 d	3.5×10^{-6}	2.92×10^{-10}
S ³⁵	10^{-6}	87 d	2.34×10^{-5}	1.95×10^{-8}
A ⁴¹	5×10^{-7}	1.8 h	2.37×10^{-8}	9.87×10^{-12}
Ca ⁴⁵	3×10^{-8}	164 d	5.23×10^{-6}	1.31×10^{-9}
Fe ⁵⁹	1.5×10^{-8}	45 d	2.03×10^{-6}	2.54×10^{-10}
Co ⁶⁰	10^{-8}	5.2 y	8.81×10^{-4}	7.3×10^{-7}
Sr ⁹⁰	2×10^{-10}	28 y	4.99×10^{-3}	8.3×10^{-10}
Y ⁹¹	4×10^{-8}	57 d	4.24×10^{-5}	1.41×10^{-9}
Ru ¹⁰⁶	3×10^{-8}	1.0 y	2.95×10^{-4}	7.4×10^{-9}
I ¹³¹	5×10^{-9}	8.05 d	8.15×10^{-6}	3.4×10^{-11}
Cs ¹³⁷	2×10^{-7}	33 y	1.26×10^{-2}	2.1×10^{-6}
Ba ¹⁴⁰	6×10^{-8}	13 d	1.37×10^{-5}	6.8×10^{-10}
Ce ¹⁴⁴	7×10^{-9}	290 d	3.10×10^{-4}	1.81×10^{-9}
Po ²¹⁰	7×10^{-11}	138.4 d	2.22×10^{-4}	1.30×10^{-11}
Ra ²²⁶	8×10^{-12}	1,620 y	1.02	6.8×10^{-9}
U	1.7×10^{-11}	4.51×10^9 y	2.86×10^6	0.04
Pu ²³⁹	2×10^{-12}	2.4×10^4 y	16.2	2.7×10^{-8}

* Values taken from "Maximum Permissible Amounts of Radioisotopes in the Human Body and Maximum Permissible Concentrations in Air and Water," NBS Handbook 52, GPO, March, 1953.

† Values taken from Kinsman, S. (ed.), "Radiological Health Handbooks," U.S. Public Health Service, 1956.

out to be only 0.00017 μ g. However, the room will contain activity amounting to 75,000 disintegrations/min, and the contamination can be readily assayed by standard methods of air sampling and counting.

SPECIAL PROBLEMS OF RESEARCH LABORATORIES

The basic techniques of radiation safety are similar regardless of the nature of the work being done. However, the practical application of those techniques does vary considerably. Research laboratories have very special problems because of the diversity and constantly changing nature of their activities. These activities must be made safe but with a minimum of interference with the exploration of new ideas and methods. In the paragraphs that follow, most aspects of health physics will be discussed with special emphasis on the needs of the research laboratory.

Basic Policies. A successful health-physics program should be based on a clear understanding of policy and of the assigned mission. As with other safety programs, basic responsibility for safety should be placed on those doing the work, i.e., on management and supervision for planning properly and providing needed facilities and on the individual workers for complying with the prescribed safety rules. The health-physics group usually has these specific duties:

1. To provide specialized advice and assistance to those doing the work
2. To provide certain assigned services
3. To keep management adequately informed of radiation safety conditions throughout the laboratory

Thus, the health-physics mission is a subtle mixture of service and staff functions.

The Health-physics Program. Figure 12-1 shows the various activities connected with radiation safety in chart form. An evaluation of radiation hazards is fre-

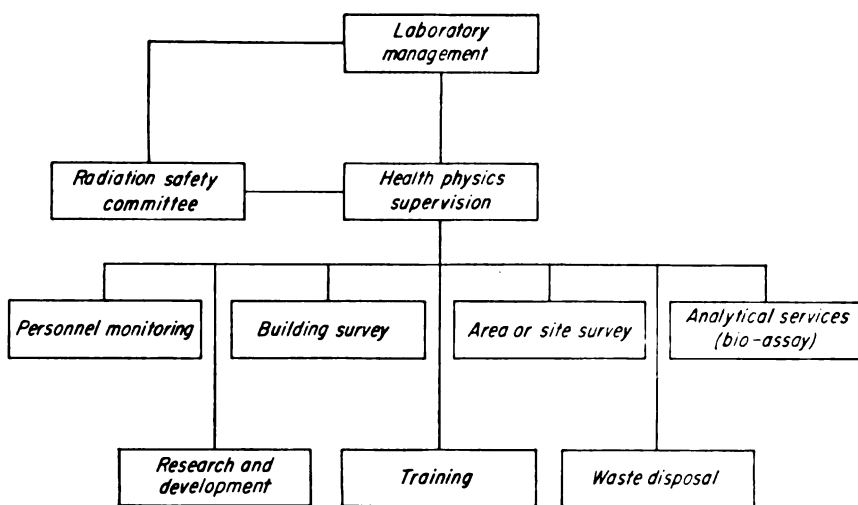


FIG. 12-1. The health-physics program.

quently referred to as a survey. Such surveys divide naturally into those concerned with conditions in a laboratory (**building survey**) and those concerned with conditions in the environs (**area or site survey**). **Personnel monitoring** refers to the routine determination of individual exposures. The term **bio-assay** has been widely used to designate the radiochemical techniques used for determining the amount of radioactive substances deposited in the body. Meteorological and geological factors are important considerations for both area survey and waste disposal; so that these disciplines, although not shown in Fig. 12-1, will usually be represented.

It is common practice to have a special committee for the purpose of establishing policies in regard to radiation safety and reviewing accidents or malpractices, if such occur. It is frequently called the radiation safety committee or the isotope committee. For small laboratories, the functions of this committee may be discharged by a single individual.

TECHNIQUES OF RADIATION PROTECTION

The basic method used to achieve safety in the use of radiation and of radioactive materials depends on three factors; determination of the radiation levels or contamination levels involved, evaluation of the meaning of those levels in terms of previously established safe standards of exposure, and the use of whatever techniques are necessary to get the work done safely. A large part of the balance of this section deals with such techniques, either by themselves or in connection with the special problems of high-voltage accelerators and nuclear reactors. Let us now consider each of the principal techniques of radiation protection individually, starting with protection by time limitation.

Protection by Time Limitation. The exposure accumulated during work in a radiation field depends both on the radiation level and on the time spent by an individual at the point in question. Thus, it is possible to limit exposures by limiting the time during which exposure is received. Table 12-2 shows how this method of

Table 12-2. Time to Accumulate an Exposure of 50 or 300 Mrms

Dose rate	Time	Dose, mrem
6¼ mrem/hr	8 hr	50
50 mrem/hr	1 hr	50
300 mrem/hr	10 min	50
600 mrem/hr	5 min	50
3 rem/hr	1 min	50
6 rem/hr	30 sec	50
18 rem/hr	10 sec	50
7½ mrem/hr	40 hr	300
300 mrem/hr	1 hr	300
1.8 rem/hr	10 min	300
3.6 rem/hr	5 min	300
18 rem/hr	1 min	300
36 rem/hr	30 sec	300
108 rem/hr	10 sec	300

protection works out in practice. It is possible to do a great many things in a few seconds if the work is properly planned, the equipment is in working order, and the people involved have had sufficient previous practice. Such practice can usually be obtained in a so-called "dry run" during which the high radiation levels are not present. Many jobs can be done in this way with a minimum of inconvenience and indeed sometimes with less exposure than would result from the use of clumsy remote-handling arrangements.

The removal of irradiated samples from a research pile often presents a problem amenable to solution by this technique. Samples irradiated in a complicated special apparatus provided for maintaining their temperatures at specified values may sometimes be extricated and transferred to lead containers even though levels as high as 100 rem/hr are encountered at a point 2 ft from the apparatus where the

worker must stand momentarily. Numbers similar to those shown in Table 12-2 should be firmly fixed in a health-physics surveyor's mind so that he manages his time in high-level locations almost intuitively.

Protection by Increased Distance. Many radiation sources are relatively small physically so that the radiation levels produced vary inversely as the square of the distance. Thus, very substantial reductions in exposure can be realized by increasing the distance between the source and the worker. This fact is the basis for the extensive use of remote-handling and remote-control arrangements for radiation work. Table 12-3 shows how this works out for Co-60, a material frequently used for gamma-ray sources.

Table 12-3. Cobalt-60 Exposures

Distance	Time for 50 mr exposure *									
	1,000 c	100 c	10 c	1 c	100 mc	10 mc	1 mc	100 μc	10 μc	1 μc
1 cm						1.4 s	14 s	2.2 m	22 m	3.7 h
1 in.					0.9 s	8.8 s	1.5 m	15 m	2.5 h	25 h
10 cm				1.4 s	14 s	2.2 m	22 m	3.7 h	37 h	
1 ft			1.3 s	13 s	2.1 m	21 m	3.5 h	35 h		
1 m		1.4 s	14 s	2.2 m	22 m	3.7 h	37 h			
10 ft	1.3 s	13 s	2.1 m	21 m	3.5 h	35 h				
10 m	14 s	2.2 m	22 m	3.7 h	37 h					
100 ft	2.1 m	21 m	3.5 h	35 h						

* The above table is based on a dose rate of 1.33 r/hr at 1 m from a 1-curie source. Air absorption is neglected since it will not be appreciable for the range of distances covered. Times are given in seconds (s), minutes (m), or hours (h).

Protection by Shielding. Under many circumstances the protection provided by time limitation and by working at a distance is not sufficient and the radiation

levels must be reduced by the use of shielding. It is important to bear in mind that particles such as alphas, betas, and protons can be completely stopped by a thickness of shielding determined mostly by their energy, whereas X- and gamma radiations are progressively reduced but never completely stopped.

Alpha particles from naturally radioactive materials are stopped by the dead outer layers of the skin and thus are unimportant for external exposures. They are very damaging when deposited in the body where they can irradiate sensitive tissues such as the bone marrow.

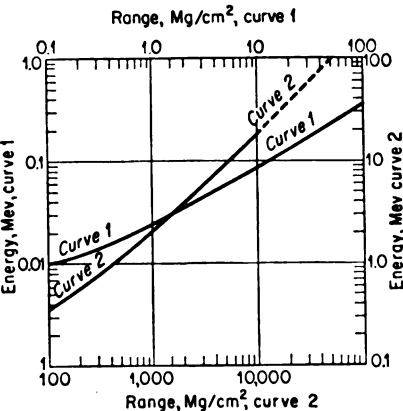


FIG. 12-2. Range of electrons in aluminum.

Beta particles are given off by a great many radioactive isotopes, and a knowledge of their range is very important in connection with safety. Values of range in aluminum can be deduced from the following equations (Katz, L., and A. S. Penfold, *Revs. Modern Phys.*, vol. 24, p. 28, January, 1952) and are shown in the graph of Fig. 12-2.

$$R \text{ (mg/cm}^2\text{)} = 412 E_0^{(1.265 - 0.0954 \ln E_0)} \quad 0.01 \leq E_0 \leq 2.5 \text{ Mev}$$

$$R \text{ (mg/cm}^2\text{)} = 530 E_0 - 106 \quad 2.5 < E_0 < 20 \text{ Mev}$$

where R = range, mg aluminum/sq cm

E_0 = energy of incident electrons

These values of range refer to the most energetic particles if the maximum energy of the beta emitter is utilized. The range data in mg/cm² for aluminum can be used

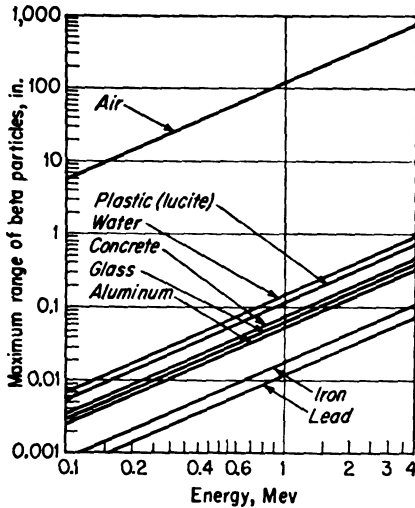


FIG. 12-3. Range of electrons in various materials. (From "The Industrial Uses of Radioactive Fission Products," Stanford Research Inst. Rept. 361.)

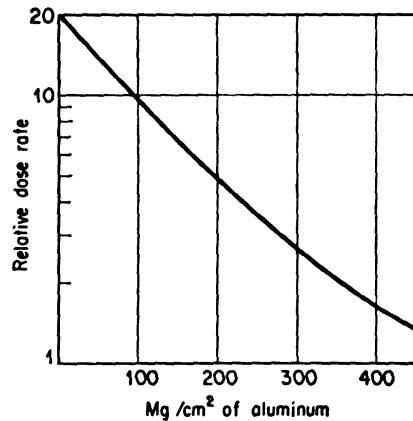


FIG. 12-4. Relationship between dose rate and absorber thickness for P-32 beta rays.

in practice with relatively little error for other light elements or for tissue. For heavy elements, ranges will be somewhat greater, however, since there are progressively fewer electrons per nucleon as the atomic number is increased and there is some decrease in stopping power per electron. Values of range in inches will depend on the density of the material. Data for some common shielding materials are shown in Fig. 12-3.

Since a great many of the beta particles from a given isotope are of lower energy, dosage rates will be progressively reduced as the shield is increased from zero up to the value corresponding to the range of the most energetic particles. Figure 12-4 shows the relationship between dose rate and absorber thickness for P-32, a beta emitter with a maximum energy of 1.71 Mev. The range of 1.71-Mev electrons is 790 mg/cm², but we note that the dose rate is reduced to one-tenth of the initial

value by an absorber of only 360 mg/cm² (unpublished data of Haughey, F. J., and L. C. Barcus, Brookhaven National Laboratory). Additional information on the range of charged particles will be presented below in connection with the discussion of high-voltage accelerators.

Monoenergetic gamma rays limited to a narrow beam are attenuated according to the following exponential law:

$$I = I_0 e^{-\mu_0 x}$$

where I_0 = incident gamma-ray intensity

I = gamma-ray intensity after passing through an absorber

μ_0 = linear absorption coefficient

x = thickness of absorber

It is important to emphasize that the attenuation of gamma rays by shields can be computed with this simple equation only for narrow-beam conditions where there is negligible scattered radiation arriving at the point of measurement. When there is a broad beam of gamma rays, for instance, behind a shield surrounding an intense source, a great deal of scattered radiation is received from the shield, and the resulting attenuation is far less than would be computed from the simple exponential law just given. This situation is taken care of analytically by the use of "build-up factors."

Under some circumstances complicated mixtures of gamma rays and X-rays are encountered, for instance, in the case of mixed fission products. Under such circumstances the low-energy components will be removed first and the average energy or hardness of the beam will increase as the shield is penetrated. A more extensive treatment of the theory of gamma-ray shielding and tables of absorption coefficients and build-up factors will be found in Secs. 7 and 8.

Protective Clothing. Protective clothing is worn by radiation workers to keep radioactive substances from coming in contact with the body or from contaminating personal clothing. For most low-level work a laboratory coat is standard, but for work at high levels or with hazardous substances, more complete protection is required. Special shoes or shoe covers, coveralls, head covers, and gloves are widely used and are similar to those used for protection in nonradioactive operations.

While lead-loaded aprons and gloves are effective against diagnostic X-rays, most gamma rays are so penetrating that it is not practical to use garments to shield individuals. In the case of beta rays, the shielding effect of garments may be appreciable. For instance, two layers of cloth, such as is commonly used for coveralls, reduced the beta dosage from uranium by 22 per cent, according to one author (Rothenberg, S. A., *Coverall Shielding to Beta Radiation*, *AEC Rept. NYO-1543*).

Contaminated garments are laundered and then reused. The degree of contamination that can be tolerated without provision of special controls or safety features for the laundering operation has not yet been determined. It is known that grossly contaminated garments require such special precautions and that very slightly contaminated ones pose no hazard.

Respiratory Protection and Air Sampling. Protection of workers against the inhaling of radioactive substances is particularly important since once material is deposited in the body it is difficult to assay and often slow to be eliminated.

Respiratory protection, when viewed from a broad point of view, consists not

only of devices such as respirators and masks worn by the individual, but also of air sampling necessary to evaluate the hazards and air-cleaning or ventilation arrangements.

Devices worn by workers are similar to those commonly used for protection against nonradioactive toxic dusts, vapors, and gases. These include respirators, face masks, supplied-air masks, ventilated suits, etc. The situation is similar in regard to air sampling except that the dust collected is assayed for radioactivity by means of a suitable counter. The reader is referred to Sec. 20 for a more detailed discussion of respiratory protection and air-sampling techniques and to the following selected list of references on the subject.

Morgan, G. W., and C. R. Buchanan, Air Contamination and Respiratory Protection in Radioisotope Work, *AEC Rept. AECU-2821*, Jan. 19, 1953.

Harris, W. B., H. D. LeVine, and M. Eisenbud, Field Equipment for the Collection and Evaluation of Toxic and Radioactive Contaminants, *AMA Arch. Ind. Health*, vol. 7, p. 490, June, 1953.

Friedlander, S. K., L. Silverman, P. Drinker, and M. W. First, "Handbook on Air-cleaning Particulate Removal," Department of Industrial Hygiene, School of Public Health, Harvard University (available from U.S. Superintendent of Public Documents).

Control of Surface Contamination. In most laboratories considerable effort is devoted to detecting and controlling surface contamination. Detection is by means of a suitable instrument held near the surface in question or by counting a "wipe" or "smear" obtained by rubbing a paper disk on the surface in a roughly standardized fashion. When surface contamination becomes high enough, it may be a health hazard in that radioactive material may be ingested or inhaled by workers. However, contamination interferes seriously with experiments, and levels are often kept far below those dictated by health considerations for this reason.

Standards for surface contamination vary from place to place, and it is not possible, at present, to establish engineering data for safety in this connection.

H. F. Henry and coworkers have made studies aimed at establishing relationships between the degree of contamination and the levels found on clothing, between clothing levels and inhalation, between hand contamination and ingestion, etc. (Bailey, J. C., and R. C. Rohr, Air-borne Contamination Resulting from Transferable Contamination on Surfaces, and The Inhalation of Radioactive Materials as Related to Hand Contamination, *Repts. K-1088 and K-1071, K-25 Plant, Carbide and Carbon Chemicals Co., Oak Ridge, Tenn.*). Such relations are, of course, highly dependent on local conditions and the work habits of those concerned.

It is usual to ban eating and smoking in areas where such activities might result in ingestion of dangerous amounts of radioactive substances. Great care should be exercised to keep contamination out of cuts and sores as transfer to the blood stream is very efficient in such cases.

Personnel Monitoring. At first glance, it might seem that personnel monitoring, i.e., the measurement of the exposures actually received by individuals, could not be considered to be a protective measure since the data are obtained after the exposure has been incurred. However, since exposure limits are now established in terms of averages over a 3-month period, values of weekly or biweekly exposures are very useful in revealing faulty practices and permitting corrective action to be taken before any harm has been done.

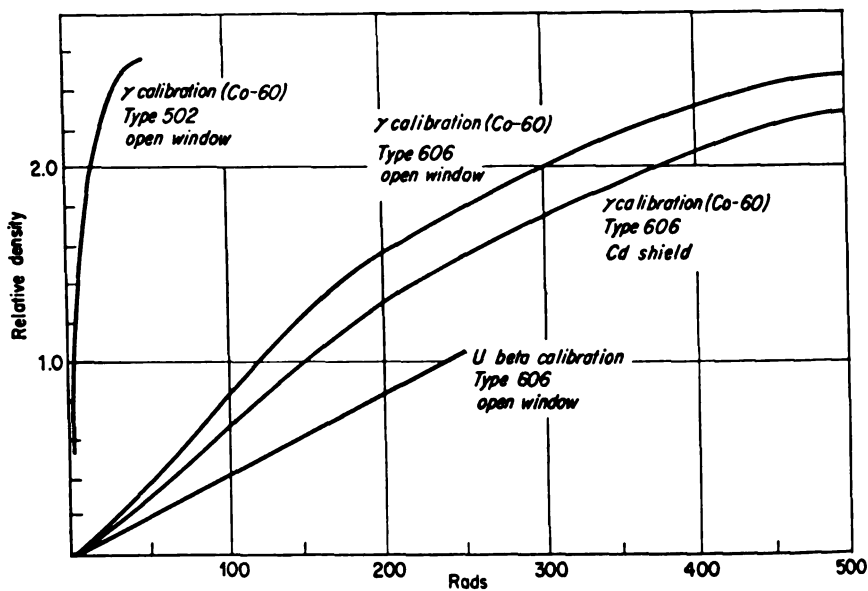


FIG. 12-5. Calibration curves for wide-range personnel-monitoring film dosimeter. (From Cheka, J. S., *Nucleonics*, vol. 12, no. 6, p. 40, June, 1954.)

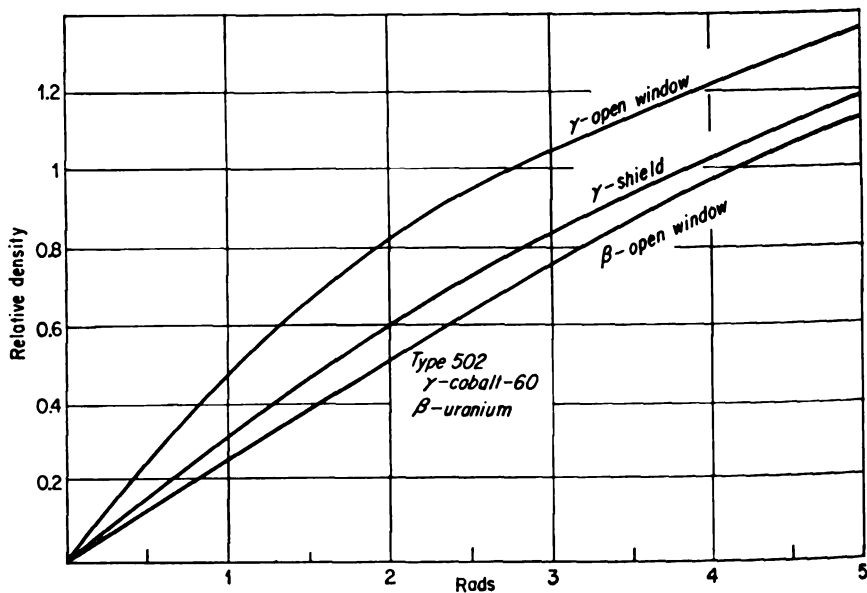


FIG. 12-6. Calibration curves for personnel-monitoring film dosimeter in low-exposure region.

Self-reading pocket dosimeters, usually of the quartz-fiber variety, are very useful for keeping track of exposures on a current basis when working in high fields of radiation. However, a **film dosimeter**, or as it is sometimes called, a film badge, is usually provided to obtain a permanent record of individual exposure.

A variety of film dosimeters are in use for personnel monitoring. Perhaps the commonest uses a dental-size film packet in a holder that has an open window at one end and a cadmium or lead filter at the other. The open window is provided to permit response due to beta rays, and the filter is provided to correct the very high response of the film to X-rays and give a measurement of X- and gamma-ray dose. In order to provide the range needed for accidental overexposures as well as the sensitivity needed for ordinary exposure levels, a packet containing two films is used. Figure 12-5 shows the relationship between density relative to an unexposed film and dose, for a Du Pont type 559 packet containing type 502 sensitive film and type 606 insensitive film.

For most films, even when worn for several weeks, a calibration curve covering the range from 0 to 5 rads or less will suffice. Such a calibration is shown in Fig. 12-6. It should be emphasized that the sensitivity of films changes somewhat from batch to batch even for the same emulsion and that it is well to do a fresh calibration curve for use with each new batch of film. For best results, one or two calibration films should be developed with each group of films being processed. A relative density of 0.01 is about the lower limit of practical readability for most densitometers, and this corresponds to about 30 mrad of gamma radiation using the shielded portion of the film. If we include other errors, we see that the uncertainty of reading small exposures amounts to ± 50 mrad with this particular packet.

This film badge has several defects. For instance, the cadmium filter leaves an undesirable peak in the response in the X-ray region. Lead is considerably better in this respect if the proper thickness for the particular film being used is chosen. Because of the effect of the paper wrapping, the beta response of a film packet drops off sharply for beta emitters of less than 0.8 Mev energy (Storm, E., *The Response of Sensitive 552 Dupont Film to Beta Radiation*, *Los Alamos Sci. Lab. Rept. LA-1284*). The most serious defect is that there is no way to differentiate between beta-ray exposures and X-ray exposures, both of which will produce heavy darkening of the unshielded portion of the film.

In order to get an idea of the energy of X- or gamma radiation and differentiate the resulting exposure from exposure due to beta rays, a dosimeter with multiple filters is required. Such a dosimeter has been developed by the health-physics group at Oak Ridge National Laboratory (Davis, D. M., E. D. Gupton, and J. C. Hart, *Applied Health Physics Radiation Survey Instrumentation*, *Oak Ridge Natl. Lab. Rept. ORNL-332*, 1st rev.). Calibration curves are shown in Fig. 12-7. It is seen that, by comparing densities under the various filters, a rough idea of the effective energy of the electromagnetic radiation can be obtained. Beta rays will record through the thinnest filter but will be stopped by the others. Thus, small X-ray exposures will not be mistaken for large beta exposures as is likely to be the case for the dosimeter with a single filter. A somewhat different dosimeter based on the same principle has been described by Tochelin (Tochelin, E., R. H. Davis, and J. Clifford, *A Calibrated Roentgen-ray Film Badge Dosimeter*, *Am. J. Roentgenol. Radium Therapy*, vol. 64, p. 475, September, 1950).

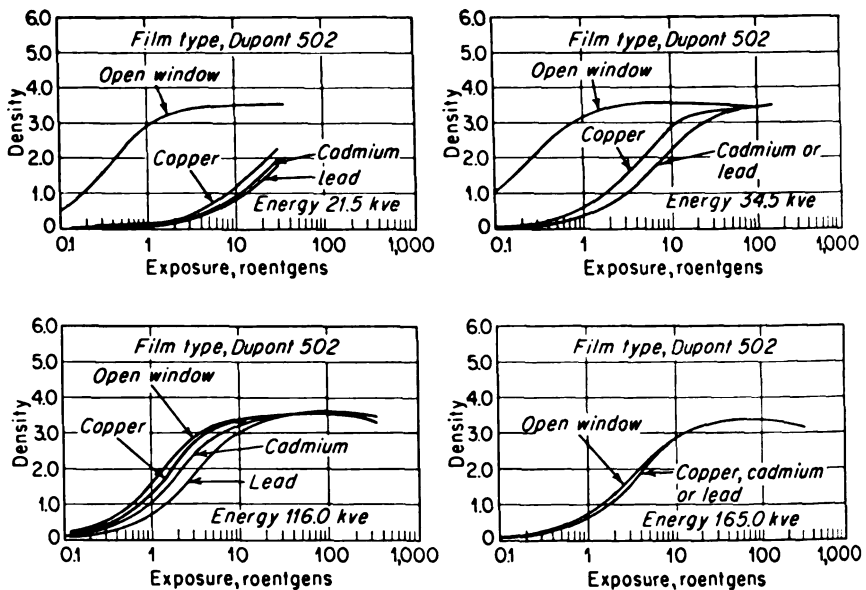


FIG. 12-7. Calibration curve for ORNL film dosimeter. (From Oak Ridge Natl. Lab. Rept. ORNL-332, 1st rev.)

It should be emphasized that film-dosimeter results are not completely definitive even with the more complex type of badge and that an investigation should be made of all sizable exposures. The information obtained as a result of this investigation combined with the film results will usually make possible an accurate evaluation of the exposure.

The detailed procedures used in personnel monitoring will vary considerably, depending on local conditions and the scale of operations. The techniques used at one large site have been described in detail (Craft, H. R., J. C. Ledbetter, and J. C. Hart, Personnel Monitoring Operating Techniques, Oak Ridge Natl. Lab. Rept. ORNL-1411).

For individuals who must receive exposures amounting to a sizable fraction of the maximum permissible amount, it is desirable to maintain a 13-week running average so that the status of each is known. In keeping individuals informed of their exposure status, it is convenient to use a number which includes all components of exposure and always has the same maximum permissible value per week. For this purpose, this author has used a number referred to as the "exposure index" whose maximum permissible value at present is 300 per week. In accordance with present standards for external exposure ("Permissible Dose from External Sources of Ionizing Radiation," NBS Handbook 59, GPO, September, 1954) this is the sum of the X- or gamma exposure in milliroentgens plus the beta exposure in millirads divided by 2 or 5 depending on whether the beta radiation is hard or soft, plus other exposures, such as those to neutrons, in millirems.

Neutron Personnel Monitoring. Monitoring of personnel for neutron exposures is important in the case of reactors and accelerators. At present, the only available

method is the nuclear-track system developed in this country by Cheka [Cheka, J. S., Neutron Monitoring by Means of Nuclear Track Film (NTA), *Oak Ridge Natl. Lab. Rept.* 547]. The method consists in counting tracks produced in a fine-grain emulsion by the passage of neutrons. Kodak NTA film is commonly used for this purpose, and the tracks are counted with a high-power (980 $\times$) microscope using an oil-immersion objective. Considerable variation in technique is possible, depending on the sensitivity desired and the type of equipment. One method is to count discrete fields while another is to count tracks in a series of continuously scanned strips. Details of the method, as applied at Hanford, will be found in Watson (Watson, E. C., Fast Neutron Monitoring of Personnel, *Rept.* HW-21552, Hanford Works, General Electric Co.).

The NTA film records the passage of fast neutrons as a result of the proton recoils that they produce. Since about 350 kev of proton energy is required to produce a 3-grain track, neutrons below about $\frac{1}{2}$ Mev are not recorded. Thermal neutrons produce tracks as a result of the Nitrogen n - p reaction. The energy response of this method is shown in Fig. 12-8, which is taken from a recent paper by Cheka (Cheka, J. S., Recent Developments in Film Monitoring of Fast Neutrons, *Nucleonics*, vol. 12, no. 6, p. 40, June, 1954). It matches the desired biological-response curve well up to $3\frac{1}{2}$ Mev but reads high above that energy. Cheka has designed a special packet in which this defect is corrected by including a suitable aluminum filter to cut down the number of proton recoils coming into the film from the paper wrapping that surrounds it.

The nuclear-track method suffers from two major defects that go hand in hand. The sensitivity is low, and scanning the films requires considerable time on the part of skilled technicians. From 300 mrems of Po-Be neutrons the result is only 30 tracks in 25 fields of 0.15 mm diameter. A tenfold increase can be realized by means of the strip-scanning technique but with a greater expenditure of time, especially where short tracks are numerous, as for a fission spectrum.

Since the number of tracks observed is small for exposures well below the maximum permissible limit, the statistical uncertainties are large. However, average exposures over a 13-week period will be more reliable. When interpreting the results of NTA-film neutron monitoring, it is important to keep in mind the wide variations to be expected because of the statistical uncertainties just referred to. Table 12-4 shows the probability of obtaining various track counts when the average for an infinite number of counts would be 10 tracks. Background fog due to gamma-ray

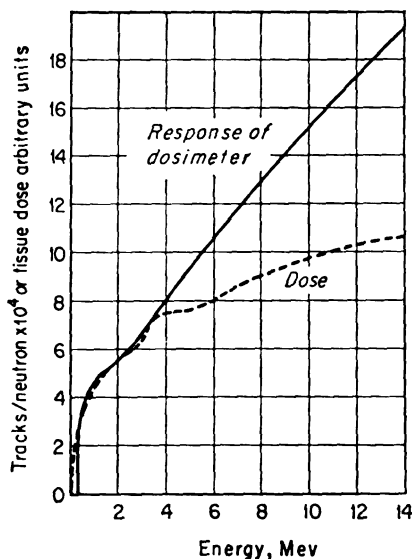


FIG. 12-8. Energy response of neutron dosimeter using NTA nuclear-track film.

exposure serves to mask neutron tracks, and about 1 r of such exposure will make track counting very inaccurate or impossible. Smaller amounts of gamma exposure will result in the counting of a few tracks not caused by neutrons if the technician is doing a good job of counting short tracks, as must be done to keep as low an energy threshold for detection as possible. This spurious counting arises from the chance juxtaposition of fog grains in a pattern identical with a short proton-recoil track.

Table 12-4. Probability of Observing Various Proton Track Counts When the Average of an Infinite Number of Counts Is 10

<i>No. of tracks</i>	<i>Probability, %</i>
4	0.5
5	1.8
6	4.1
7	7.1
8	9.9
9	11.9
10	12.5
11	11.9
12	10.5
13	8.6
14	6.6
15	4.9
16	3.4
17	2.9
18	1.5
19	1.0
20	0.6

Evaluation of Radiation Levels. A knowledge of radiation levels is fundamental to planning and executing work safely. Thus, the measurement of radiation levels with survey meters is properly included as one of the techniques of radiation protection. Such instruments are discussed in detail elsewhere in this volume, but it is well to emphasize the necessity for having instruments that will give the surveyor

as complete information as possible about the radiation being studied. For instance, a thick-walled ionization chamber is fine for a Cobalt-60 source where only gamma radiation is encountered but would be entirely unsuited for use with P-32 or mixed fission products where beta dosage is an important factor.

In estimating radiation levels in advance, the surveyor determines the sources of radiation to be expected and determines the nature of the radiation from tables of isotopes, data on neutron activation, past experience, etc. Figure

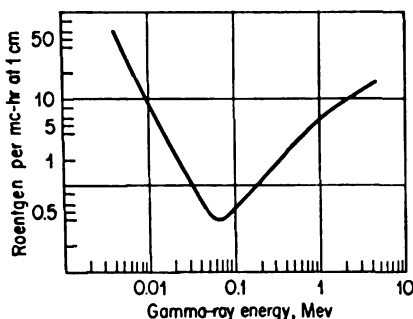


FIG. 12-9. Dose-rate energy relationship for gamma emitters.

12-9 is very useful in estimating gamma-ray levels when the number of curies of emitter is known. Care must be taken to determine the decay scheme of the emitter and to add emission rates for gammas that are in cascade or to average properly those which arise from alternate modes of decay. In many cases, preliminary estimates will be very crude or in error so that actual measurements of radiation levels should be made when the experiment is carried out. A conservative approach is proper when doing a new experiment for the first time. On the basis of the survey data, precautions often can be relaxed subsequently with safety.

Many laboratories provide specially trained health physicists to assist in evaluating radiation hazards. However, in many cases the experimenters can make their own measurements more efficiently and should be encouraged to do so.

Bio-assay. The estimation of the body burden of internal radioactive contaminants by analysis of the excreta is quite commonly referred to as bio-assay. This term is correctly used to designate the process of assaying certain substances by observing the biological effects that they produce but has been appropriated for the above purpose for lack of a better term.

Internally deposited radioactive isotopes are excreted from the body via the urine, feces, perspiration, and breath. The breath is used to evaluate the amount of radium in the body, but for most other substances urinalysis is the preferred method. Having determined the daily excretion rate of a particular activity, metabolic data are used to estimate the quantity present in the body, and the result is compared with the maximum permissible values given in NBS Handbook 52.

Bio-assay, like personnel monitoring, may be classed as a protective technique as it enables one to detect internal contamination before it becomes serious and to take corrective steps. Most health physicists feel, however, that it should be a second line of defense and that the primary effort should be directed toward preventing the entrance of radioactive substances into the body. This feeling arises from a number of difficulties inherent in bio-assay procedures. In the first place, it is not easy to obtain reliable 24-hr samples from the individuals involved. In the second place, most of the chemical procedures are involved and require great skill in tracer chemistry. In the third place, the biological data needed for an accurate evaluation of body burden are often crude or even lacking entirely and the limits set in NBS Handbook 52 suffer from the same uncertainties. Despite these difficulties, bio-assay is still an important protective technique since, despite the best efforts of all concerned, workers do occasionally get significant amounts of activity into their bodies.

It is important to establish a fixed policy in regard to interpretation of the results of urinalysis. This author uses the following somewhat arbitrary scheme.

1. Consider excretion rates of less than 10 per cent of the maximum permissible value to be negligible.
2. Consider excretion rates between 10 and 25 per cent of maximum permissible to be cause for mild concern and corrective action.
3. Consider excretion rates between 25 and 50 per cent of maximum permissible to be cause for serious concern and strong corrective action.
4. Consider excretion rates of over 50 per cent of maximum permissible as cause for transfer to work where absolutely no more activity can enter the body until the excretion rate drops below the 25 per cent level.

Unfortunately, no published schedule of maximum permissible urinary excretion rates exists at present principally because of the lack of necessary data on the ratio of urinary to fecal elimination rates.

The substances for which analyses are probably most frequently made in urine are strontium, uranium, plutonium, polonium, and tritium, but analyses have been worked out for many others. An analysis for gross activity that assays most of the elements in the middle of the periodic table including the fission products but excludes the urinary potassium is convenient for screening groups of people. There is a considerable literature on this subject to which the reader is referred for details. A selected list of references follows.

McClelland, J., Analytical Procedures of the Industrial Hygiene Group, *Los Alamos Sci. Lab. Rept.* LA-1858.

Shubert, J., Estimating Radioelements in Exposed Individuals, *Nucleonics*, vol. 8, no. 2, p. 13, February, 1951; no. 3, 66, March, 1951; no. 4, 59, April, 1951.

Mawson, C. A., and I. Fischer, The Estimation of Radioactive Strontium and Other Fission Products in Urine and Water, *Natl. Research Council Can. Rept.* CRM-455.

Campbell, E. E., B. M. Head, and M. F. Milligan, An Extraction Method for the Determination of Uranium Alpha Activity in Urine, *Los Alamos Sci. Lab. Rept.* LA-1920.

Farabee, L. B., Procedure for the Radiochemical Analysis of Strontium and Barium in Human Urine, *Oak Ridge Natl. Lab. Rept.* ORNL-1932.

Hempelmann, L. H., and W. H. Langham, Determination of Systematically Deposited Plutonium in Laboratory Personnel and a Simple Qualitative Test for Exposure to Airborne Radioactive Material, *Los Alamos Sci. Lab. Rept.* AECU-2633.

Aerico, J., S. Foti, and J. Fast, Fluorimetric Uranium Determination in Urine, *AEC Rept.* 4, Health and Safety Division, New York Operations Office.

Proceedings of Bio-assay and Analytical Chemistry Meeting, Oct. 6 and 7, 1955, Health and Safety Division, National Lead Co. of Ohio.

SAFETY PROBLEMS OF A RESEARCH REACTOR

Reactor Accidents. Many safety features of a research reactor are inherent in the design, and it is in the design phase that the groundwork for future safe operations must be laid. One basic problem is that of guarding against an accident of some kind that might release huge quantities of radioactive materials and cause serious hazards in the surrounding regions. This problem has been discussed in Sec. 15 and is very well covered in the following references.

Teller, E., *et al.*, "The Essentials of the Evaluation of Reactor Hazards," Paper 8/P/853, International Conference on the Peaceful Uses of Atomic Energy, Geneva, August, 1955.

Parker, H. M., and J. W. Healy, "Environmental Effects of a Major Reactor Disaster," Paper 8/P/482, International Conference on the Peaceful Uses of Atomic Energy, Geneva, August, 1955.

Fitzgerald, J. J., H. Hurwitz, Jr., and L. Tonks, Method for Evaluating Radiation Hazards from a Nuclear Incident, *Knolls Atomic Power Lab. Rept.* KAPL-1045, March, 1954.

In general, one can say that the objective should be to make the reactor as safe as feasible in regard to its time constant, excess reactivity available, temperature coefficient, etc., and to provide multiple automatic safety instrumentation to pro-

tect against human error. It is present practice to isolate reactors from populated areas or to enclose them in shells that will contain the fission products in case of accidents. A great deal is being learned about the seriousness of reactor accidents from the planned power excursions carried out at the AEC reactor test station in Idaho (Dickson, J. J., *et al.*, "Experimental Determinations of the Self-regulation and Safety of Operating Water-moderated Reactors—Results and Motion-picture Records," Paper 8/P/481, International Conference on the Peaceful Uses of Atomic Energy, Geneva, August, 1955), and from a recent theoretical study (Theoretical Possibilities and Consequences of Major Accidents in Large Nuclear Power Plants, *AEC Rept. WASH-740*, March, 1957).

Safety Features of Reactor Design. In the paragraphs that follow, we shall discuss only safety problems associated with normal reactor operations. Several design features are important in this connection:

1. The basic shielding of the reactor should reduce neutron and gamma-ray fluxes to a low value such that individuals will not receive an appreciable fraction of the maximum permissible dose as a result of radiation escaping through the shield.
2. Plugs and shutters for holes in the basic shield should be of a design to minimize or eliminate leakage of radiation, particularly of neutrons.
3. Placement of holes from which external neutron beams may be obtained should, wherever feasible, be carefully chosen to pose the minimum difficulty in protecting personnel from such beams.
4. Alarms actuated by continuous monitors should be planned for warning of unusual radiation levels.
5. Adequate space should be provided for issuing and collecting protective clothing and for decontaminating equipment and personnel.
6. Provision should be made for the office and equipment of the health-physics staff.

Reactor Start-up. During start-up and initial operation of a new reactor, a detailed study of the efficiency of the shielding and an evaluation of all possible radiation hazards should be made. Extreme caution should be exercised during this period since the staff is inexperienced and the hazards not yet fully evaluated. An initial survey at reduced power is desirable. Consideration should be given to the possibility of leaks caused by voids in the shield, poorly fitting plugs, scattering around the edges of an inadequate shield, or by diffusion along structural members penetrating the shielding material.

Whenever changes are made in experimental holes or apparatus using beams from the reactor, or any other work has been done that might result in leaks, a check of the areas concerned should be made prior to start-up and a radiation survey should be made immediately after the reactor comes up to operating power. Occasional surveys are necessary, even for reactors where no known changes in the shield have been made, since it is possible for unsuspected changes to occur, particularly in the case of a research reactor.

Neutron Beams. Neutron beams brought out of reactors should be provided with protective arrangements that will prevent overexposure of personnel. Since the dose rate in such beams will usually be hundreds or thousands of rems per hour, the beam should usually be enclosed in a pipe or fenced off and terminated in a substantial catcher. Less intense beams may be roped off or merely protected by

warning signs. Experimental materials placed in neutron beams by experimenters may result in objectionably high levels of scattered neutrons in the vicinity. These may sometimes be reduced by changes in the apparatus. If not, local shielding may be necessary or access to the area involved may be controlled.

Since the eyes are especially sensitive to fast-neutron irradiation, exposure of the lenses of the eyes to fast neutrons must be carefully limited to the prescribed 300 mrem/week. Beams are of special concern in this connection, and it is essential to keep people from sighting along beams from reactors even for a few seconds.

Experience has shown that the neutron fluxes from many reactors are such that the fast-neutron hazard heavily outweighs that due to thermal neutrons. For instance, Marley and Smith (Marley, W. G., and B. S. Smith, "Control of Radiation Hazards in the Operation of Medium-powered Experimental Reactors," Paper 8/P/452, International Conference on the Peaceful Uses of Atomic Energy, Geneva, August, 1955) quote levels of 1,000, 200, and 10 rem/hr respectively due to fast neutrons, slow neutrons, and gamma rays in a beam from the British research reactor BEPO. The situation is similar with the Brookhaven reactor, not only in beams but also in locations receiving only scattered radiation as a result of the use of external beams. Thus, it is of primary importance to have survey instruments for evaluating fast-neutron fluxes and personnel-monitoring equipment responding to fast neutrons.

Radiation Monitors for Reactors. Because of the varied activities being carried out by a large number of people, it is desirable to equip a research reactor with auto-

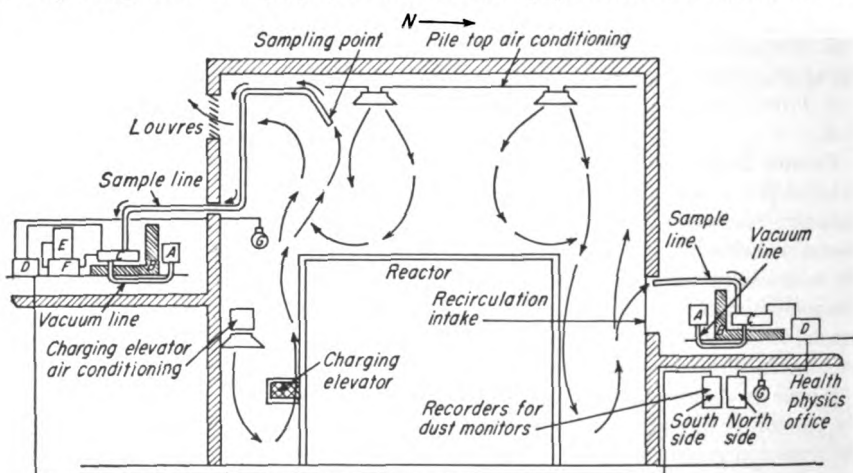


FIG. 12-10. Health-physics air-monitoring system, schematic arrangement. (A) Vacuum pump. (B) Lead shield. (C) Filter-paper transport. (D) Beta-gamma rate meter. (E) Recorder. (F) Alpha scaler. (G) Air-contamination alarm.

matic alarms to warn of high radiation levels or high levels of atmospheric contamination. For monitoring gamma levels, ionization chambers or scintillation-type meters, placed at strategic locations and set to sound an alarm at some level well above normally expected values, are used. Each unit can be self-contained, or remote indication at a central location can be provided.

The degree of atmospheric contamination likely to be encountered will depend strongly on the type of pile and experiments in progress. Such contamination may result from any substance that has been irradiated in the pile and is especially likely to occur during charging or discharging fuel or as a result of maintenance activities. Figure 12-10 shows an arrangement that has been very successful at the Brookhaven reactor. The air samplers are of the moving-filter-strip type with the beta-gamma counters arranged to detect activity as soon as it is collected. The collecting points are such that atmospheric contamination released anywhere in the vicinity of the reactor will be detected.

Reactors are, in general, designed to cause rather modest disturbances to natural background during normal operations. However, it is desirable to maintain some surveillance of radiation conditions outside the reactor building and in neighboring areas so that unusual situations can be properly evaluated. With this objective in mind, it is proper to determine background radiation levels prior to the start-up of a new reactor. After start-up, the same equipment may be used to evaluate the effect of the reactor. Continuous monitoring of reactor effluents is desirable, both as a means of estimating the probable effect on the environment, at least on a relative scale, and as a means of quickly detecting difficulties.

Miscellaneous Operating Problems. In view of the fact that on start-up of a reactor radiation levels increase strongly, an aural warning system should be installed to ensure that all personnel in the vicinity of the reactor are made aware that start-up is contemplated. Personnel should be able to communicate with the reactor operator, in the event that the start-up is unsafe, within a period of time that is less than the warning period. Throughout the building in which the reactor is housed, there should be continual visual indication to inform personnel as to whether the reactor is in operation or not.

Administratively it is desirable that changes in apparatus or shielding be made only with the approval of operations or radiation safety personnel specifically designated to be responsible for such action. Since, in the event of an accident, it may be necessary to evacuate the building quickly, a plan should be devised for this evacuation and responsible persons should be designated and available at all times for its execution.

One of the most difficult problems encountered with a research reactor is that of removing equipment from experimental holes and performing various necessary operations on it. If properly designed, such equipment will, in so far as possible, use materials known to be as little activated by neutron irradiation as possible. Even such well-designed equipment will nevertheless be extremely radioactive and requires special techniques for removal. A waiting period after shutdown for decay of the shorter-lived activities is very helpful if the nature of the experiment will permit. In some cases, it is possible to pull the apparatus or sample out of a region of intense flux into one of relatively low flux near the outside of the pile prior to shutdown and obtain a considerable decay in addition to that available during the shutdown period. Figure 12-11 shows such a removal operation in progress. One individual is winding up a wire on a shielded windlass and thereby drawing a very active apparatus from the pile into a shielded container. A second is removing as much loose contamination as possible with a vacuum cleaner while the third is monitoring the radiation level. A certain amount of contamination will inevitably get

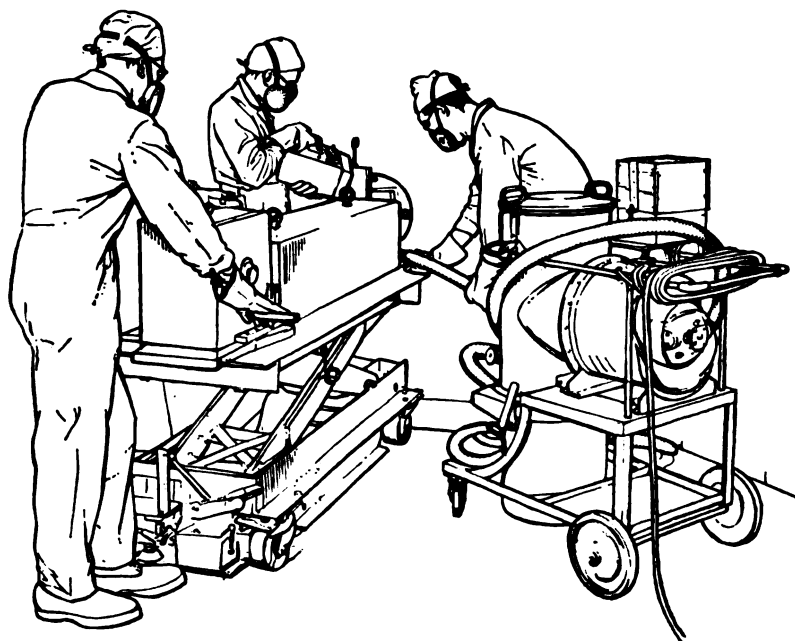


FIG. 12-11. Removing radioactive apparatus from an experimental hole in a research reactor.

onto the floor; so the area has been isolated and will be cleaned as soon as the removal is completed. In other cases the apparatus may be pulled into a wooden box which can be quickly pulled on a trailer to an area where it can be left for further decay or where special facilities for working on it have been set up.

ACCELERATOR SAFETY

High-voltage accelerators are extensively used in research laboratories and pose some of the most serious hazards and most difficult control problems. In the majority of cases the machine is enclosed in a vault which adequately shields it during operation. However, beams are frequently brought out through the shield, and in some cases, machines have been built with only partial shields or no shield at all. The conditions of use of accelerators and the personnel using them change frequently, making safety even more difficult.

The hazards encountered will, of course, vary widely depending on the type of machine. Electron accelerators will produce principally gamma radiation while proton or deuteron accelerators will produce neutrons whose number and energy will depend on the type of target used. Maintenance work on the machine or demounting targets may be difficult because of high gamma- and beta-radiation levels. An external beam is likely to be extremely hazardous both because of the high dose rates in the actual beam and because of radiation scattered by apparatus placed in the beam. Accelerators of a few hundred million electron volts or more produce a

Table 12-5. Ranges of Particles in Air

Energy, Mev	Electrons		Protons		Deuterons		α particles		μ mesons	
	G/cm ²	M	G/cm ²	M	G/cm ²	M	G/cm ²	M	G/cm ²	M
1	0.41	3.34	0.0028	0.023	0.002	0.017	0.0006	0.0050	0.0135	0.110
2	0.95	7.75	0.0087	0.071	0.006	0.046	0.0012	0.0100	0.046	0.375
5	2.6	21.2	0.042	0.34	0.025	0.208	0.0042	0.034	0.238	1.94
10	5.2	42.4	0.141	1.14	0.083	0.68	0.0130	0.106	0.82	6.7
20	10.5	85.8	0.485	3.96	0.282	2.30	0.042	0.34	2.78	22.7
50			2.52	20.6	1.45	11.8	0.208	1.70	12.6	103
100			8.70	71	5.03	41.0	0.72	5.90	35.3	288
200			29.2	238	17.4	142	2.58	21.0	89	730
500			132	1,080	85.3	696	13.0	106	251	2,050
1,000			364	2,980	263	2,140	42.9	350	500	4,080
2,000			895	7,350	731	5,960	132	1,080	970	7,900
5,000			2,520	20,600	2,350	19,200	490	4,000	2,220	18,100
10,000			5,040	41,100	5,030	41,000	1,180	9,600	4,150	33,800
20,000									7,700	63,000
50,000									17,400	142,000

complex mixture of radiations including mesons and alpha particles. Table 12-5 gives values of range in air for a number of particles over a wide range of energies.

The following is a list of the chief factors having to do with accelerator safety:

1. The original design should provide adequate protection for experimenters in the form of either shielding or arrangements for remote operation.

2. Arrangements should be made for exclusion of people from certain areas where the levels are dangerously high, and these arrangements should be as fool-proof as possible.

3. Suitable warning should be provided when the machine is to be turned on, and there should be an indicator to show when the machine is operating.

4. Clearly marked emergency switches should be provided so that a person locked into a high-level area can turn the machine off.

5. Responsibility for safety checks should be clearly fixed. In general, it is better to depend on previously established procedures and on regular operators than on informal checks by experimenters.

6. Radiation alarms are desirable in areas where unexpectedly high levels may occur.

7. Radiation surveys should be made in detail for each new type of operation. Quick checks will suffice for future work of the same kind.

8. Adequate survey instruments should be provided for evaluating the X- and gamma-beta and neutron dose rates in millirems per hour.

9. Personnel-monitoring equipment should be provided to measure the exposures actually received by personnel.

10. Necessary remote-handling equipment and local shielding arrangements needed for maintenance work and target handling should be provided.

11. In some cases surface or air-borne contamination will be a problem. Equipment for evaluating these hazards and protecting against them should be available.

12. Personnel using the accelerator should be indoctrinated in regard to the hazards involved and the techniques required for safety.

Not all these factors will apply to each accelerator, of course, and in particular, some of the smaller machines, or ones restricted to a few modes of operation, will be fairly easy to operate safely.

Safety features of different machines are so diverse that a detailed general description is not feasible. However, most of the principles enumerated above can be illustrated by discussing the Brookhaven 60-in. cyclotron, whose safety features are shown in Fig. 12-12.

All the **interlocks** shown are arranged to turn off the r-f oscillator (S8) if opened and thus interrupt the acceleration of particles. The recordlock (S1) is a device that records the time and key number whenever turned on. There are only four keys, and they are assigned to specific responsible operators. Thus, responsibility for safe operation is clearly established. This cyclotron is provided with a separate vault or tunnel for experiments with the external beam. There is a photoelectric barrier (S5) across the entrance to this area, since for some experiments, personnel need to work in the tunnel but out of the actual beam region. In such cases, the tunnel-door interlock is shorted out at the console with interlock control switch S6.

When there is no external beam, the tunnel is safe and the photoelectric barrier can

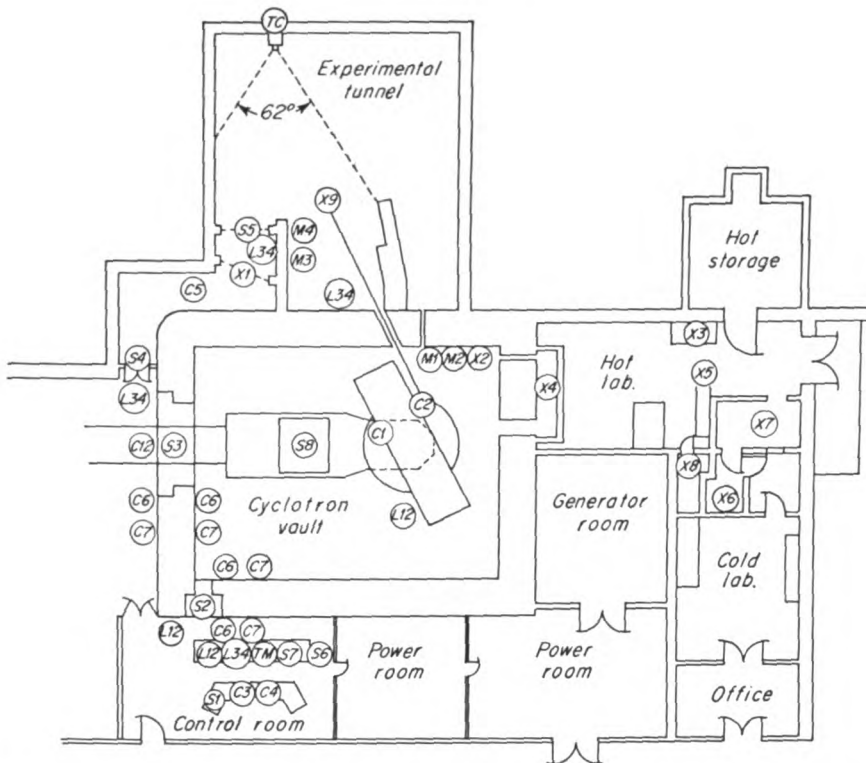


FIG. 12-12. Cyclotron safety features. *Controls:* (C1) Internal interceptor probe. (C2) External interceptor probe. (C3) Switch for internal interceptor probe. (C4) Switch for external interceptor probe. (C5) Switch for interceptor probes. (C6) Door buttons. (C7) Emergency door lowerer. *Monitors:* (M1) Gamma-vault monitor—detector. (L12) Gamma-vault monitor—light. (M2) Neutron-vault monitor—detector. (L12) Neutron-vault monitor—light. (M3) Gamma-tunnel monitor—detector. (L34) Gamma-tunnel monitor—light. (M4) Neutron-tunnel monitor—detector. (L34) Neutron-tunnel monitor—light. (TC) Television camera. (TM) Television monitor. *Interlocks:* (S1) Recordolock. (S2) Little door. (S3) Big door. (S4) Tunnel door. (S5) Photoelectric barrier. (S6) Interlock control switch (door to experiment tunnel). (S7) Interlock control switch (photoelectric barrier). (S8) α -f oscillator. *Miscellaneous:* (X1) Photoelectric warning barrier. (X2) Vault bell. (X3) Fume hood. (X4) Dry box. (X5) Walk-in hood. (X6) Shower room. (X7) Locker room. (X8) Toilet. (X9) Target location.

be shorted out with interlock control switch S7. Such interlock control switches provide the operating flexibility necessary in a research machine.

Both the cyclotron vault and the experiment tunnel are provided with **gamma and neutron monitors** (M1, M3, M2, M4). Ionization chambers are used as gamma detectors and are set to operate warning lights above a level of 5 mr/hr. Scintillation detectors of the zinc sulfide in lucite variety are used as fast-neutron detectors and are set to operate at fluxes above about 30 neutrons/cm²/sec. Lights are shared

by the gamma and neutron monitors, with a flashing light indicating gamma radiation and a steady light indicating both gamma and neutron radiations. The lights illuminate signs reading "beam on."

A closed-circuit **television monitor** in the control room permits the operator to inspect the beam area before turning on the beam. This monitor is very useful since the tunnel is at some distance from the control room and contains much apparatus requiring the attention of experimenters. The main cyclotron vault is personally inspected by the operator before the small door (S2) is closed.

Interceptor probes are provided for cutting off either the internal or external beams without otherwise interfering with the adjustment of the machine. Switches (C3, C4, C5) for controlling these probes are provided as shown. The door-opening button (C6) inside the vault will stop the operation of the machine quickly by means of the door interlock. The vault door can be lowered mechanically (C7) if the electric mechanism fails.

Most of the miscellaneous items listed in Fig. 12-12 are self-explanatory. A warning bell in the vault rings continuously while the vault door is being closed. A second photoelectric barrier (XI) ahead of the interlocked one (S5) is provided to ring a bell and prevent time-consuming interruptions in the smooth operation of the cyclotron caused by forgetful or overeager workers.

The new **billion-volt accelerators** pose a variety of new problems in both dosimetry and control. The general principles are the same, but radically different arrangements are likely to be necessary. For instance, it may not be feasible to shield the area for external-beam experiments, but instead the beam may be suitably terminated and a large fenced-in area provided. In this case, everyone except experimenters will be excluded from the fenced-in area, and the access of the experimenters to the really hazardous parts of the experimental area will be circumscribed.

A number of accelerators have been surveyed recently by the New York Operations Office of the AEC and the results reported in the following document (Solon, L. R., J. E. McLaughlin, and H. Blatz, *Stray Radiation Measurements at Particle Accelerator Sites*, *AEC Rept.* NYO-4699, June 1, 1956). While this report refers to specific accelerators, it is a good source of ideas in regard to safety problems of accelerator design and operation.

In conclusion, one can hardly overemphasize the importance of carefully and formally analyzing the hazards associated with accelerators and of providing adequate protective arrangements. Experience has shown that most unsafe practices arise from human carelessness or indifference or from human fallibility in operating even a nominally safe machine.

General References

- Katz, L., and A. S. Penfold, *Revs. Modern Phys.*, vol. 24, p. 28, January, 1952.
Unpublished data of Haughey, F. J., and L. C. Barcus, Brookhaven National Laboratory.
Rothenberg, S. A., Coverall Shielding to Beta Radiation, *AEC Rept.* NYO-1543.
Bailey, J. C., and R. C. Rohr, Airborne Contamination Resulting from Transferable Contamination on Surfaces, and The Inhalation of Radioactive Materials as Related to Hand Contamination, *Repts.* K-1088 and K-1071, K-25 Plant, Carbide and Carbon Chemicals Co., Oak Ridge, Tenn.

Storm, E., The Response of Sensitive 552 Dupont Film to Beta Radiation, *Los Alamos Sci. Lab. Rept.* LA-1284.

Davis, D. M., E. D. Gupton, and J. C. Hart, Applied Health Physics Radiation Survey Instrumentation, *Oak Ridge Natl. Lab. Rept.* ORNL-332, 1st rev.

Tochelin, E., R. H. Davis, and J. Clifford, A Calibrated Roentgen-ray Film Badge Dosimeter, *Am. J. Roentgenol. Radium Therapy*, vol. 64, p. 475, September, 1950.

Craft, H. R., J. C. Ledbetter, and J. C. Hart, Personnel Monitoring Operating Techniques, *Oak Ridge Natl. Lab. Rept.* ORNL-1411.

"Permissible Dose from External Sources of Ionizing Radiation," NBS Handbook 59, GPO, September, 1954.

Cheka, J. S., Neutron Monitoring by Means of Nuclear Track Film (NTA), *Oak Ridge Natl. Lab. Rept.* 547.

Watson, E. C., Fast Neutron Monitoring of Personnel, *Rept.* HW-21552, Hanford Works, General Electric Co.

Cheka, J. S., Recent Developments in Film Monitoring of Fast Neutrons, *Nucleonics*, vol. 12, no. 6, p. 40, June, 1954.

Teller, E., *et al.*, "The Essentials of the Evaluation of Reactor Hazards," Paper 8/P/853, International Conference on the Peaceful Uses of Atomic Energy, Geneva, August, 1955.

Dickson, J. J., *et al.*, "Experimental Determinations of the Self-regulation and Safety of Operating Water-moderated Reactors—Results and Motion-picture Records," Paper 8/P/481, International Conference on the Peaceful Uses of Atomic Energy, Geneva, August, 1955.

"Theoretical Possibilities and Consequences of Major Accidents in Large Nuclear Power Plants," AEC, Washington, D.C., March, 1957.

Marley, W. G., and B. S. Smith, "Control of Radiation Hazards in the Operation of Medium-powered Experimental Reactors," Paper 8/P/452, International Conference on the Peaceful Uses of Atomic Energy, Geneva, August, 1955.

Solon, L. R., J. E. McLaughlin, and H. Blatz, Stray Radiation Measurements at Particle Accelerator Sites, *AEC Rept.* NYO-4699, June 1, 1956.

Selected Reading

Morgan, G. W., and C. R. Buchanan, Air Contamination and Respiratory Protection in Radioisotope Work, *AEC Rept.* AECU-2821, Jan. 19, 1953.

Harris, W. B., H. D. LeVine, and M. Eisenbud, Field Equipment for the Collection and Evaluation of Toxic and Radioactive Contaminants, *AMA Arch. Ind. Health*, vol. 7, p. 490, June, 1953.

Friedlander, S. K., L. Silverman, P. Drinker, and M. W. First, "Handbook on Air-cleaning Particulate Removal," Department of Industrial Hygiene, School of Public Health, Harvard University (available from U.S. Superintendent of Public Documents).

McClelland, J., Analytical Procedures of the Industrial Hygiene Group, *Los Alamos Sci. Lab. Rept.* LA-1858, 2d ed.

Shubert, J., Estimating Radioelements in Exposed Individuals, *Nucleonics*, vol. 8, no. 2, p. 13, February, 1951; no. 3, p. 66, March, 1951; no. 4, p. 59, April, 1951.

Mawson, C. A., and I. Fischer, The Estimation of Radioactive Strontium and Other Fission Products in Urine and Water, *Natl. Research Council Can. Rept.* CRM-455.

Campbell, E. E., B. M. Head, and M. F. Milligan, An Extraction Method for the Determination of Uranium Alpha Activity in Urine, *Los Alamos Sci. Lab. Rept.* LA-1920.

Farabee, L. B., Procedure for the Radiochemical Analysis of Strontium and Barium in Human Urine, *Oak Ridge Natl. Lab. Rept.* ORNL-1932.

Hempelmann, L. H., and W. H. Langham, Determination of Systematically Deposited Plutonium in Laboratory Personnel and a Simple Qualitative Test for Exposure to Airborne Radioactive Material, *Los Alamos Sci. Lab. Rept.* AECU-2633.

Aerico, J., S. Foti, and J. Fast, Fluorimetric Uranium Determination in Urine, *AEC Rept.* 4, Health and Safety Division, New York Operations Office.

Proceedings of Bio-assay and Analytical Chemistry Meeting, Oct. 6 and 7, 1955. Health and Safety Division, National Lead Co. of Ohio.

Teller, E., *et al.*, "The Essentials of the Evaluation of Reactor Hazards," Paper 8/P/853, International Conference on the Peaceful Uses of Atomic Energy, Geneva, August, 1955.

Parker, H. M., and J. W. Healy, "Environmental Effects of a Major Reactor Disaster," Paper 8/P/482, International Conference on the Peaceful Uses of Atomic Energy, Geneva, August, 1955.

Fitzgerald, J. J., H. Hurwitz, Jr., and L. Tonks, Method for Evaluating Radiation Hazards from a Nuclear Incident, *Knolls Atomic Power Lab. Rept.* KAPL-1045, March, 1954.

Section 13

MEDICAL RADIATION APPLICATIONS

By

JACK S. KROHMER, B.S., M.A., FACR (Assoc.), *Assistant Professor of Radiology, University of Texas Southwestern Medical School, Dallas, Texas; Chief Physicist, Parkland Memorial Hospital, Dallas, Texas; Consultant Physicist, Veterans Administration Hospital, Dallas, Texas; Physicist, The Radiation Center, Fort Worth, Texas; Consultant in Radioisotopes, U.S. Naval Hospital, Portsmouth, Virginia; Diplomate of American Board of Radiology in Radiological Physics.*

General.....	13-2
X-ray Procedures.....	13-3
X-ray Diagnosis.....	13-4
X-ray Therapy.....	13-11
Radioactive Materials in Medicine.....	13-18

MEDICAL RADIATION APPLICATIONS

Jack S. Krohmer

GENERAL

See page 13-28 for General References.

Radiation sources are used in the medical field in three major categories, namely, (1) diagnosis, (2) therapy, and (3) research. Diagnosis includes X-ray diagnostic procedures in addition to the relatively new diagnostic studies which are carried out with radioactive isotopes. Therapy includes X-ray therapy procedures, use of radium and similar materials, and the more recent uses of artificially produced radioisotopes. Included in the category labeled research are applications of all types of medical radiation sources in which the procedure is considered to be in the experimental or investigative stage.

In the use of medical radiation sources there arise a number of **protection problems**. The first of these is human protection, which must be considered from three standpoints: (1) the protection of hospital or office personnel, (2) nonoccupational protection of humans, and (3) the protection of patients during the various radiation procedures. It is obvious in the case of personnel that we must protect all persons working for prolonged periods in close proximity to radiation sources; thus great emphasis is placed upon this phase of the protection problem. The **protection of medical personnel** (Ferlazzo *et al.*, Radiation Hazard Evaluation and Control in Hospitals, *Radiology*, vol. 65, pp. 892-902, 1955) includes some problems which are unique to the medical radiation field. In the first place, in a well-rounded medical radiation department, the personnel are exposed to a very wide variety of radiations: for example, very soft radiations from diagnostic X-rays, more penetrating radiations from therapeutic X-rays, very penetrating radiations from radium, and also alpha and beta radiations from radioactive materials. This situation makes the problem of monitoring the dosage to these personnel extremely difficult, since the response of the various monitoring devices may differ greatly depending upon the type of incident radiation (see Sec. 3 on Exposure Standards and Sec. 23 on Personnel Control Measures). Medical personnel are not under as close control as are the personnel concerned with radiation sources in industry and other related fields, and thus the possibility of overexposure to radiation is much greater. There is an especially great potential danger to the hands through radiation received during fluoroscopy, holding of patients during diagnostic procedures, and in the handling of radium, during both preparation and introduction into the patient. There is also a possibility of contamination and subsequent introduction of radioactive materials

into the bodies of medical personnel during their use and handling of radioactive materials. Although the **protection of patients** is more difficult, every effort should be made to maintain the lowest possible radiation exposure to the patient while still obtaining the results desired in the radiation procedure.

A second type of radiation protection which is important in the medical field is the **protection of undeveloped film** which will be used, or which has been used, in diagnostic studies of patients. Ordinarily, the protection required for film (barriers and distance) will have to be greater than that required to protect human beings, since film may be fogged and thus made unusable by very small amounts of radiation (as low as 0.15 mr/hr) (Wilsey, R. B., *The Use of Photographic Films for Monitoring Stray X-rays and Gamma Rays*, *Radiology*, vol. 56, p. 229, 1950). This aspect of radiation protection is quite often overlooked, thus making necessary very costly remodeling procedures which must be carried out in order to afford proper film protection.

As a result of the present rather widespread use of radioactive materials and the associated **counting equipment**, a new phase of the protection problem has come about. This protection must be applied in order to keep the background levels of counters and other such equipment down to usable values.

X-RAY PROCEDURES

X-rays are defined as any electromagnetic or photon radiation which is emitted from points outside of atomic nuclei. A more practical and fully adequate definition from the standpoint of this book is that X-rays are electromagnetic or photon radiation emitted from the target of electron accelerators, including X-ray tubes. It is important to note that the radiation emitted from these sources is emitted in a continuous spectrum of energies. Even though one may have an X-ray unit which operates at a constant potential of 200,000 volts, the X-rays emitted will vary continuously in energy from 0 to 200,000 ev. Further modification of the energies available in an X-ray beam is brought about by choice of the waveform of the potential applied to the X-ray tube. These variations include the use of half-wave and full-wave rectified potentials, both biased and unbiased, and constant potentials (see also Sec. 6).

For the purposes of protection, X-rays emitted by usual X-ray units may be classified as (1) the **useful beam**, which is that beam emitted through the orifice provided in the X-ray tube enclosure; (2) the **leakage radiation**, which is the radiation emitted through all parts of the tube enclosure other than the orifice; and (3) the **scattered radiation**, which includes all radiation resulting from scattering of the useful beam by any material which it may strike. Because of the high intensity of the emitted radiation, and also because of the fact that the radiation generators and installations are generally quite large in size, X-ray protection problems are more complex than those associated with some of the other medical radiation sources. X-ray protection problems are somewhat lessened by the following: this type of radiation source may be turned off; the useful beam is well collimated and thus may be limited in size and direction; and also, these units are generally operated from remote-control panels.

A first step in the **protection against X-radiation** may be achieved by a restriction

of the beam orientations and a limitation of beam size. The best and most economical (providing space is available) type of protection from X-radiation, as well as other types of radiation, is the provision of distance between the source and occupied areas. Distance attenuation of radiation follows the **inverse-square law** and in addition provides some attenuation of the beam through air absorption. Ordinarily, the above protective measures are not adequate to protect personnel, film, etc., completely against the effects of X-rays. Therefore, we must rely upon the interposition of barriers between the source and the occupied area. According to the presently recognized terminology, **primary barriers** must be provided for any position which the useful beam of radiation may strike, and **secondary barriers** must be provided at all other positions against leakage and scattered radiation (see also Sec. 8). A very complete coverage of the formulas, curves, and tables necessary in the determination of barrier thicknesses for various medical X-ray units is included in NBS Handbook 60. This handbook should be referred to for all such barrier calculations. It is worthy of mention here that lead provides the best barrier against low-voltage or low-energy X-rays (250 kv or less) because of its high density, high atomic number, and ease of procurement. As one goes to higher energies, concrete and other building materials become relatively better than lead. Beyond an energy of 1 Mev concrete and the like become the materials of choice for radiation protection. This relative increased effectiveness of concrete and similar materials is brought about by a decrease in the importance of photoelectric absorption at the higher X-ray energies (see Sec. 7). At the higher energies, absorption of photon radiation is dependent mainly upon the density of the absorbing material, and thus we find that concrete is about one-fifth as effective as lead for a given thickness of material. At the lower X-ray energies, namely, those produced at about 100 kvp, concrete is approximately one-eightieth as effective as Lead (see Sec. 8 for further material on the principles of shielding).

X-RAY DIAGNOSIS

Types of Units. Usual X-ray diagnostic procedures are carried out in the range of 50 to 150 kvp. However, voltages as high as 1 to 2 million volts are used for some techniques. The latter supervoltage radiography techniques are presently being used in relatively few institutions and, as is obvious, make necessary more extensive protective measures than are required for the more conventional units (see Supervoltage Therapy installations below).

The **X-ray tube head**, or protective enclosure on X-ray diagnostic machines, must meet the requirements set forth by the National Committee on Radiation Protection (NCRP) in NBS Handbook 60. These requirements state that a diagnostic protective tube enclosure should limit the leakage radiation at 1 m from the X-ray target to 0.1 r/hr with the tube operating at its maximum continuous rated current and voltage. In addition, all shutters, cones, and other beam-defining devices are to include protective material so as to limit the radiation to the same levels as the tube enclosure itself. It may be stated that most modern X-ray diagnostic protective enclosures include protection sufficient to meet or exceed these requirements.

Recent diagnostic protective rules (NBS Handbook 60) also state that all fixed diagnostic X-ray equipment should include a **minimum filtration** of 2.5 mm (portable

diagnostic units require 1.5 mm Al) of aluminum or equivalent. This means that, if the X-ray tube provides an inherent filtration equivalent to 0.5 mm of aluminum, an additional 2 mm is to be added in the path of the useful beam. This filtration has the effect of minimizing the dose to the entrance surface of the patient while affecting, to only a minor degree, the already greatly attenuated radiation which reaches the film or the fluoroscopic screen.

Radiographic installations are those in which X-radiation is used to produce an image on photographic film. Because of the relatively short times of actual X-ray use of radiographic equipment, protection is generally not too great a problem. Tables giving the barriers required for radiographic units at various kilovoltages and distances have been made up and are available in NBS Handbook 60. (Note: addendum of Jan. 8, 1957, requires the addition of 1.6 HVL and 5.0 HVL respectively for controlled and uncontrolled areas. These additions must be made to the barriers listed in the tables of Handbook 60, 1955 edition.) Special consideration should be given to the fact that the useful radiographic beam will be oriented in many directions and thus may strike many more positions or walls than is generally the case for therapeutic equipment. Orientations are usually limited to those below the horizontal, and thus radiographic protection against the useful beam need extend only to a height of about 7 ft above the floor. Secondary barriers are to be provided for other wall and ceiling areas.

A major problem associated with the radiographic use of X-ray is that of **immobilization of the patient**, especially small children. Although many immobilization devices have been produced, a large number of patients must be hand-held. As is obvious, such holding of patients by X-ray personnel would expose them to objectionable amounts of radiation. Because of this, it is generally recommended that, if patients are to be held, the person doing the holding should be a parent, friend, or other person who is not usually exposed to radiation. These persons are to be protected as much as is possible and are to be warned to stay out of the useful X-ray beam during the holding procedure. All other persons are to stay out of the X-ray room or behind barriers during radiographic exposures, and the patient should be observed through a protected viewport. The useful beam should always be limited to as small a size as is practicable for the particular procedure involved. This limiting may be accomplished by diaphragms or cones, or by the use of leaded rubber or other types of masking materials (masking materials should transmit no more than 5 per cent of the useful beam—NBS Handbook 60). Limitation of beam size will cut down considerably on the amount of scattered radiation to which the patient is exposed and will also limit the areas and volumes of the patient which are to be exposed to the useful beam. An additional hazard to the patient during certain radiographic procedures is due to the use of a compound known as **Thorotrast**, a so-called contrast medium. Thorotrast is a contrast medium containing thorium, which, because of its high atomic number, casts a dense shadow on X-ray film, thus allowing easy visualization of its area of deposition. The material is colloidal in nature and, as such, is selectively deposited to a large extent within the liver and remains entrapped in this location for a period of many years, with lesser amounts deposited in the spleen, bone marrow, and lymph nodes. This entrapped thorium, being an alpha emitter, can produce subsequent radiation effects within these tissues and has been thought to produce rather serious late effects in patients (Looney,

W. B., Late Clinical Changes Following the Internal Deposition of Radioactive Materials, *Ann. Internal Med.*, vol. 42, pp. 378-387, 1955). Because of these complications and because of the development of better alternative procedures, Thorotrast is very rarely used at present.

A **fluoroscopic X-ray installation** is one in which X-rays are used to produce a visible image on a fluorescent screen. This type of X-ray unit requires long exposure times, the average fluoroscopic examinations consuming up to 5 min of actual X-ray exposure.

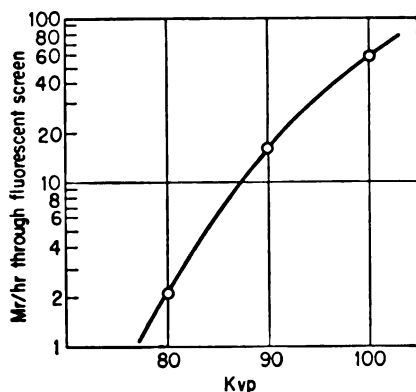


FIG. 13-1. Radiation dosage transmitted through fluorescent screen for a typical fluoroscopic X-ray unit as a function of kilovoltage applied to X-ray tube. Tube current 5 ma; inherent filtration 3 mm aluminum; X-ray target to fluorescent screen distance 30 in.; X-ray target to measuring chamber distance 32 in. (Courtesy of E. Dale Trout, General Electric X-ray Co., Milwaukee, Wis.)

A special property of this type of installation minimizes the radiation protection problem in that the fluoroscopic screen is covered by a layer of protective leaded glass which attenuates the useful beam of radiation to relatively low levels (see Fig. 13-1). Because of this leaded-glass screen which is mounted so as to be always in the useful beam (for all modern fluoroscopic units), there need be no barrier protection against the useful beam, and thus all protection in fluoroscopic installations is of a secondary-barrier type.

Tests must be made, however, to see that the useful radiation beam is actually limited to the area covered by the protective leaded-glass screen. It is presently recommended that, at the maximum table-to-screen distance which will be used, the useful radiation beam with the shutters wide open shall leave a dark border of at least $\frac{1}{4}$ -in. width around the edge of the fluorescent screen

(NBS Handbook 60). This may be accomplished by limiting the opening of the shutters and also by limiting the table-to-screen distance.

Although barrier protection for fluoroscopic installations is simple, this type of unit presents probably the **greatest hazard to X-ray personnel**, since the doctor and possibly one or two helpers must be within the X-ray room during the X-ray procedure. Because of this, they will be exposed to the scattered and leakage radiation, as well as to that part of the useful beam which passes through the protective leaded-glass screen. Such exposure to X-ray personnel should be minimized by keeping the exposures as short as possible, by being sure that the physician is well dark-adapted so that he may easily view the image with a small X-ray tube current, and by having all personnel in the room at the time of fluoroscopy wear leaded rubber or other types of protective aprons and gloves.

An **image intensifier** is a device which may be appended to a usual fluoroscopic X-ray unit so as to intensify electronically the image which is produced by the X-rays (Morgan, R. H., Screen Intensification: A Review of Past and Present Research with an Analysis of Future Development, *Am. J. Roentgenol.*, vol. 75, pp. 69-76,

1956). This type of apparatus makes it possible to fluoroscope with very small X-ray tube currents. Because of this, and because of the fact that the fluoroscopic image may be viewed indirectly through a mirror at the side of the X-ray table, exposures to both patient and personnel are considerably lower than in usual fluoroscopic procedures. The use of this type of apparatus is becoming more widespread and should, in the future, eliminate or cut down considerably the hazard to personnel during fluoroscopic procedures.

Photofluorographic apparatus (Hirsch, I. S., *Fluorography, the Photography of the Fluoroscopic Image*, *Am. J. Roentgenol.*, vol. 43, p. 45, 1940) is that which produces a photograph of the image produced on a fluoroscopic screen by X-rays. This type of apparatus is generally used for examination of the chest or upper gastrointestinal tract in chest-survey installations and in many hospital admitting rooms. Occasionally, this apparatus is installed within the radiology department of a hospital. The major reason for the use of photofluorography is that the photographic reproduction can be made on small-sized film which may be conveniently stored and which is relatively inexpensive.

Since photofluorographic apparatus uses a short (36-in. or less) X-ray-tube-to-patient distance, and since the amount of exposure necessary to produce a photographic representation of the fluoroscopic image is rather great, the possible **exposure to patient and personnel** is considerably higher than that obtained when using ordinary radiographic techniques. Photofluorographic equipment is generally operated from within the room in which the unit is installed, and thus protection for personnel is generally achieved by providing a control booth within the X-ray room. This control booth should include an adequate amount of lead shielding and should provide a means for assuring that the X-ray technician remains within this control booth during exposures. This may be achieved by providing a control cord short enough to require the technician to remain behind the protective control booth during exposures. Since the equipment may be close to secretaries and other similar personnel in admitting rooms, protection for these persons must also be provided.

Operating-room X-ray installations include cystoscopic X-ray units as well as special diagnostic units used for operative radiography. This equipment is used very much in the manner that ordinary radiographic equipment is used, except for the fact that it is within the operating room. Because of this location and because of the adjacent use of various explosive anesthetic gases, this equipment should be of an approved explosionproof type (National Electrical Code, National Fire Protection Association, 60 Batterymarch St., Boston 10, Mass.). In this type of installation, X-ray exposures are often made with medical personnel within the room, and thus it is especially important that the useful X-ray beam be limited to as small a size as possible so as to minimize the possible exposure of personnel. Operating-room personnel must be acquainted with the possible hazards associated with this equipment and should be provided with a set of rules for its safe operation.

Shoe-fitting Fluoroscopes. A special type of fluoroscopy unit operating in the range from 50 to 65 kv is in rather widespread use in shoe departments and shoe stores for the expressed purpose of obtaining proper fit of shoes. The practice is especially popular for fitting of children's shoes. Although the value and advisability of using these units have been seriously questioned (Editorial, Shoe Fitting

Fluoroscopes, *J. Am. Med. Assoc.*, vol. 15, pp. 1004-1005, 1949. Moyers *et al.*, Radiation Exposure in Shoe Stores, *N.Y. State Dept. Labor Monthly Rev. Div. Ind. Hyg.*, vol. 31, pp. 33-40, 1952), their use continues and thus many city, state, and national organizations have set up recommendations for their proper use (Safety Code for the Industrial Use of X-rays, American Standards Association, 254:1, 1946, and Safe Operation of X-ray Shoe Fitting Machines, *Bull. Bur. Ind. Hyg., Dept. Public Health Welfare*, Cleveland, Ohio).

The main hazard due to this type of unit results from the **exposure of the customer** to unnecessary amounts of radiation. The possible deleterious effects of X-ray exposure to the feet have been covered by Hempelmann (Hempelmann, L. H., Potential Dangers of the Uncontrolled Use of Shoe-fitting Fluoroscopes, *New Engl. J. Med.*, vol. 241, p. 333, 1949). It has been recommended that the dose to the feet for a single exposure should not exceed 2 r and that no more than three exposures be made on any one customer in any one day, or more than 12 fittings be carried out per year. In order to achieve this level of exposure, it is recommended that a minimum of 1 mm Al (protected against damage) should be included in the useful beam of radiation, and that the voltage and milliamperage be so adjusted that the transmission through this filter be less than 24 r/min. Using these factors, the maximum exposure time for a single fitting should be limited to no more than 5 sec. It has been stated (Editorial, Shoe Fitting Fluoroscopes, *J. Am. Med. Assoc.*, vol. 15, pp. 1004-1005, 1949) that the maintenance of these factors of exposure will probably result in unsatisfactory visualization of fit from the standpoint of the salesman, customer, and relatives of the customer. Because of this, possible overexposure may result from the customer's insisting upon too many fittings, or going from one store to another, in which case all control of exposure is lost. In this situation control must be left to the customer's intelligence and discretion, and in this regard the recommendations of the American Standards Association (ASA), which are in general accord with other recommendations, state that a sign should be placed upon the unit stating: "Repeated exposure to X-rays may be harmful. Fluoroscopic examinations for shoe fitting should be limited to no more than twelve in one year."

An additional hazard from this type of X-ray unit is the **possible overexposure of the salesperson or customers** to stray radiation. It is generally recommended that the dosage due to the stray radiation be such that the maximum permissible amount cannot be exceeded at a distance of 6 in. from the case of the unit (Safety Code for the Industrial Use of X-rays, American Standards Association, 254:1, 1946). It is also true that a relatively large amount of stray radiation is emitted from the hole through which the customer's feet enter the unit, and thus this opening should be directed away from all space occupied extensively, either by customers or by sales personnel. A survey should be made of the stray radiation around such units, and proper protective measures, such as the provision of barriers, should be carried out in order to maintain safe standards.

Dose to Patient. Since the radiation dosage sustained by the patient during X-ray diagnostic procedures is incidental to the purposes of the procedure, every effort should be made to keep this dose to as low a level as possible. This is best accomplished by minimizing the number of radiographic procedures carried out on a patient, by using the smallest X-ray fields practicable for the particular procedure

involved, by using an adequate amount of shielding (a minimum of $2\frac{1}{2}$ mm of aluminum is now recommended by the NCRP), and by using the shortest exposures possible with a minimum milliamperage through the X-ray tube. An estimate of the dose to the surface of the exposed diagnostic field may be obtained by use of the nomogram in Fig. 13-2. It should be noted that this nomogram provides the air

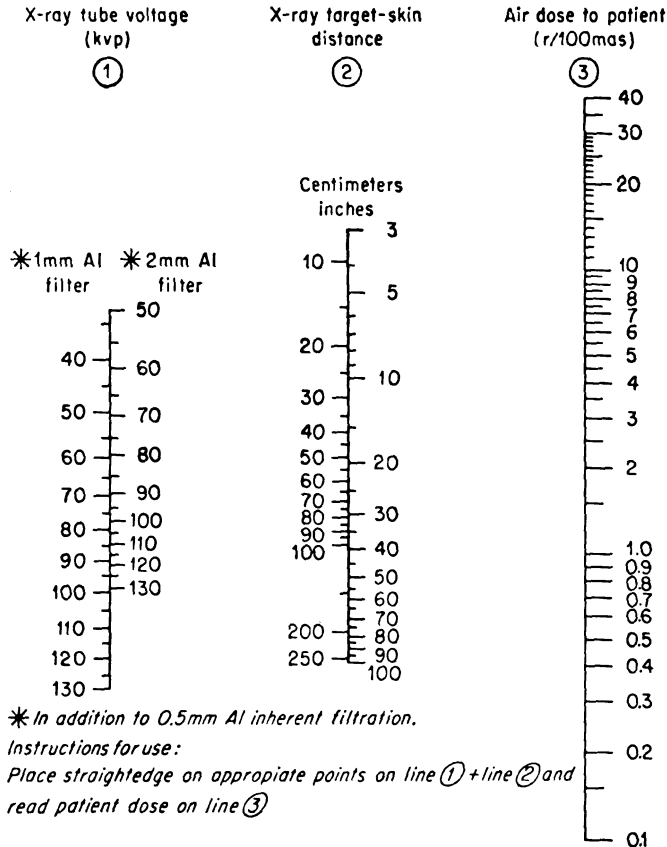


FIG. 13-2. Nomogram for determining radiation dose to the patient per 100 ma-sec of exposure during radiographic procedures. (Data for this nomogram were obtained from the paper of Sorrentino, J., and R. Yalow, *A Nomogram for Dose Determinations in Diagnostic Roentgenology, Radiology*, vol. 55, pp. 748-753, November, 1953, and from NBS Handbook 60.)

dosage to the exposed field per 100 ma-sec of radiation exposure for filtrations of both 1 mm of aluminum and 2 mm of aluminum filtration in addition to an inherent filtration of $\frac{1}{2}$ mm of aluminum. Instructions for use of the nomogram are included in Fig. 13-2.

In addition to the dose sustained by the patient in the exposed field, stray and scattered radiation dosage is also present at other areas of the patient's body during diagnostic procedures. The major area of concern for the stray and scattered

radiation dosage is the **gonads**, since very minor radiation exposure to these genetic organs may produce significant genetic changes in the offspring of persons exposed to radiation. A recent report (Laughlin, J. S., and Pullman, I., Physical Properties of Ionizing Radiations Which Affect the Population of the United States, Preliminary and Unclassified Report to the Genetics Panel of the National Academy of Sciences' Study of the Biological Effects of Atomic Radiations, Apr. 18, 1956) states that, by present standards, an estimate of the average 30-year gonadal exposure of United States citizens is about 7.4 r. Of this, about 41 per cent is due to medical examinations, 58 per cent due to natural background, and 1 per cent due to the fall-out from atomic explosions. It is clear from these figures that medical examinations account for a large portion of the gonadal exposure of the population and should be critically examined with a view toward maintaining as low an exposure as possible. A recent survey of the literature on the gonadal dose in roentgen examinations was made by Laughlin, Meurk, Pullman, and Sherman (Bone, Skin, and Gonadal Doses in Roentgen Diagnostic Procedures, *Am. J. Roentgenol.*, vol. 78, p. 961, 1957). It will be noted that there are discrepancies in the estimated gonadal dosages as shown in this report; however, this compilation represents the best available data at the present time and does indicate that the dosage to the gonads in diagnostic radiology is significant in many cases. The possible effects of the stray and scattered radiation dosage to areas of the patient, other than to the gonads, are probably quite insignificant as compared with the dose sustained by the patient in the actual exposed field. It has been recommended recently that a uniform nationwide policy be established by a group such as the American College of Radiology to keep permanent records of each patient's gonadal exposure, as well as exposures to other portions of the body. It is felt that through such a procedure some control will be achieved over the incidental exposure of the population to ionizing radiation.

Film Protection. Special attention should be given to the protection of undeveloped or unexposed film around X-ray diagnostic installations. Adequately shielded storage boxes for unexposed films should be provided at the location of each diagnostic installation as well as an adequately protected pass box which is used for transferring films from the radiographic rooms to the dark room. It should be kept in mind that the barriers provided between X-ray rooms and the dark room will ordinarily be thicker than those required for personnel protection, and will extend from floor to ceiling. Storage facilities for large quantities of unexposed X-ray film should be provided at an adequate distance from any X-ray installation. However, a small quantity of film will necessarily have to be safely stored in the dark room.

Radiation Surveys. All X-ray diagnostic installations should be subjected to a radiation survey by a qualified expert (NBS Handbook 60 and 51). A copy of the written report of this survey should be kept on file. The survey shall include a determination of the adequacy of tube-housing and beam-defining devices, measurements of the dose to patients, and measurements of stray and scattered radiation around the installation.

It should be noted that, in performing radiation surveys on diagnostic installations, care must be taken in the **choice of proper survey instruments**. A Geiger-Müller type of survey instrument can, in general, be used only as a qualitative in-

indicator of positions of possible hazard, since it does not accurately measure the radiation dosage to which a person would be exposed. This is because its response to radiation is dependent upon the energy of the radiation. A properly constructed and calibrated ionization-chamber type of survey instrument, however, will measure the dosage to which a patient may be exposed with a reasonable degree of accuracy and with a small amount of energy dependence. As a result of these facts, it is presently recommended by the NCRP that a Geiger-survey type of instrument be used to make a rapid evaluation of possible hazard and that then the actual degree of hazard be determined by measuring at all positions of possible hazard (as determined by the Geiger counter—all positions should be accurately measured where a Geiger reading of one-fifth of the permissible dosage rate was obtained) with an ionization-chamber type of survey instrument. This procedure will assure adequate protection of personnel and patients and will, in addition, provide legal protection for the organization operating the X-ray equipment.

X-RAY THERAPY

Types of Units. X-ray therapy procedures are carried out with units operating from approximately 6 kv to 2 million volts with the conventional type of therapy apparatus and are extended to 100 million volts or more with apparatus of the linear-accelerator and betatron type. In addition to kilovoltage, the waveform of the applied voltage and the filtration determine the radiation output (**quantity**) and penetrating power (**quality**) of the radiation emitted by a given therapy unit. The quantity is generally specified as exposure dose per unit time, and the quality in terms of **half-value layer** (HVL—the amount of a given material required to attenuate the useful beam by 50 per cent). Conventional therapy units are half-wave rectified, full-wave rectified, or constant-potential units. A change from half-wave rectification to full-wave rectification will result only in an increase in quantity for a given power input to the unit, whereas a constant-potential unit will provide an increase in both quantity and quality. Linear accelerators and betatrons produce a high-frequency pulsed type of acceleration with a near homogeneous energy spectrum of the electrons which subsequently produce X-ray photons (see Sec. 6).

Because the therapy patient is being purposely exposed to a relatively large amount of radiation which produces a sizable scattered dose in portions of the body other than that being treated, the **therapy protective tube enclosure** need not attenuate the leakage beam to the levels required for diagnosis. The requirement is that the enclosure and permanent beam-defining devices limit the leakage radiation to less than 1 r/hr at 1 m from the tube target with the tube operating at its maximum potential and its maximum continuous rated milliamperage. Adjustable or removable beam-defining devices and masking materials are to transmit no more than 5 per cent of the useful beam (see NBS Handbook 60).

In the field of X-ray therapy, no set **filtration** is recommended for any given unit, since filters are used to aid in increasing the quality or HVL of the radiation so as to increase the dose at a given depth in tissue. There is a maximum filter for each kilovoltage, above which little modification of penetration or quality is afforded. An increase in the filtration above this maximum value results only in a decreased radiation output of the X-ray unit with little change in penetration.

Grenz rays (Glasser, O., *The Physical Foundation of Grenz-ray Therapy, Radiology*, vol. 18, p. 713, 1932) are X-rays in the range of from 6 to 20 kvp, or from HVL of about 0.01 to 0.10 mm of aluminum. These units are used in the treatment of very superficial skin conditions, since they are almost completely absorbed before passing through the skin. Grenz rays, because of their very low penetration and small range even in air, require no protection other than that provided by 3 m of air or equivalent (see NBS Handbook 60). The main hazard from this sort of radiation is due to the useful beam, which must be limited to the particular area being treated.

A **contact-therapy unit** is one in which the X-ray tube is put in contact with, or very close to, the surface being treated (Howes, W. E., and M. R. Camiel, *Contact Roentgen Therapy, Am. J. Roentgenol.*, vol. 48, p. 360, 1942). This unit, like the Grenz-ray unit, produces its major effects on the surface of the skin and in layers slightly below this. However, in this case the decreased penetration is brought about by the fact that the X-ray tube target is extremely close to the area being treated. Thus, because of the rapid fall-off of radiation with distance, dosage levels below the skin are quite small. These units are generally operated in the kilovoltage range of from 40 to 50 kvp and with HVL of from 0.25 to 2.0 mm of aluminum. When used for true contact therapy, the useful beam is limited by a suitable cone and enters directly into the skin of the patient, and thus no useful-beam protection is needed. Scattered radiation is also down to a minimum. However, the leakage radiation at the surface of the tube housing must be kept below 0.1 r/hr since the tube is often hand-held (NBS Handbook 60). Contact units have an extremely high dose rate because of the proximity of the target to the patient, and thus a moderate distance provides adequate leakage and direct-beam protection for personnel during treatment owing to the very short exposures which are used. Accidental exposure to the useful beam at short distances, however, must be guarded against.

Superficial or low-voltage X-ray therapy installations generally are used in the range of from 60 to 150 kvp and are used at treatment distances of from about 10 to 30 cm, both with and without added filtration. The half-value layers achieved with this type of apparatus vary from 0.5 mm of aluminum to 0.5 mm of copper. The radiation produced by superficial-therapy units, without beryllium windows (see below), is very similar in quality to that produced by conventional diagnostic equipment. However, the exposure times for superficial therapy are considerably longer than those used for diagnostic equipment. Because of the longer treatment times, the protection must necessarily be somewhat more extensive than for diagnostic equipment. Although superficial-therapy units are generally used to treat conditions close to the surface of the skin, it must be realized that these radiations can be quite penetrating and can produce considerable biological effect at points quite distant from the surface of the skin, and thus patient protection must be taken into account. The superficial-therapy control panel should be outside the X-ray room, and proper barriers should be provided in the walls of the room. Although not recommended, it is permissible for units operating up to 100 kv to have the X-ray controls within the X-ray room and the operator shielded by a portable type of X-ray barrier (NBS Handbook 60).

Beryllium-window therapy installations (Cipollaro, A. C., *Beryllium Window Radiations for Superficial Therapy, Arch. Dermatol. and Syphilol.*, vol. 62, pp. 214-221, 1950) employ X-ray tubes in which the X-ray tube window is constructed of

beryllium rather than of glass. Beryllium, having a low atomic number and density, permits the emission from the tube of very low energy photons which would not be emitted through a usual glass window. Units including these tubes are used predominantly in the 20- to 100-kv range (occasionally up to 250 kvp) and have the advantage of providing an X-ray beam of very low half-value layer and consequent low penetration. These units, because of the low inherent filtration, present an extremely high dosage output with attending short treatment times which must be carefully controlled. Hazards are, for the most part, the same as for similar units having conventional glass windows. A special hazard associated with this type of unit is to the patient and is due to the possibility of treating the patient with intended added filtration accidentally omitted. This results in accidental gross overexposure of the patient, the overexposure increasing with the kilovoltage applied to the tube. The use of a filter indicator (see Patient-protection Devices in Therapy below) is extremely important.

Deep therapy includes the kilovoltage range from 150 to 400 kvp with HVL from 0.5 to 5.0 mm of copper. The radiations from these units are quite highly penetrating, producing effects at great depths within the body, and thick protective barriers are necessary. Many deep-therapy units also double as superficial-therapy units and therefore are used at different values of kilovoltage and with many different filters. Since the dosage output of these units varies considerably with the changes in kilovoltage and filtration, a safe and error-free filter indicator is needed to avoid the possibility of treating patients with modalities other than those decided upon. For all deep-therapy units, the control panel should be installed outside the X-ray room and the unit should be operated only with X-ray room doors completely closed. It is advisable to hang the doors in such a manner as to cause them to close except when opening pressure is applied.

The **supervoltage-therapy** range starts at potentials above 400 kvp and extends to 3 million volts with HVL from about 5 to 14 mm of copper. The majority of the work in this range is carried out either at 1 or 2 million volts. These voltages are, at present, usually produced by **resonance generators** (Charlton *et al.*, New Million Volt X-ray Outfit, *J. Appl. Phys.*, vol. 10, p. 374, 1939) or **Van de Graaff generators** (Trump *et al.*, Compact Supervoltage Roentgen Ray Generator using a Pressure Insulated Electrostatic High-voltage Service, *Am. J. Roentgenol.*, vol. 44, p. 610, 1940). Because of the greatly increased penetration of the radiation from the supervoltage units, barriers will be considerably thicker than for deep-therapy units. Although lead will provide the thinnest adequate barrier at this voltage range, concrete provides the most economical type of barrier. Considerable reduction in cost of shielding can be achieved by locating the supervoltage unit as far as possible from occupied areas, especially areas where counting devices are used, and quite often, earth shielding can be made use of by locating the installation below ground level. A suitably positioned maze may be used to cut down the amount of protection required in access doors. The provision of a maze requires a considerable amount of space and also makes the moving of beds into and out of the X-ray room quite difficult (NBS Handbook 60).

Supervoltage radiation, in addition to its increased penetration, possesses two **properties which make for a difference in treatment procedures and results**. The first of these is that the scattered radiation tends to be oriented predominantly in a

forward direction, thus providing a more sharply defined beam through the patient and a minimum of scattered radiation dose except at small angles of scatter. This property tends to limit the patient volume being exposed to the radiation and also tends to lessen the amount of protection required for the scattered radiation beam. The second property is that the maximum radiation dose is achieved at some distance below the surface of the patient (about 1 to 4 mm), thus making necessary the

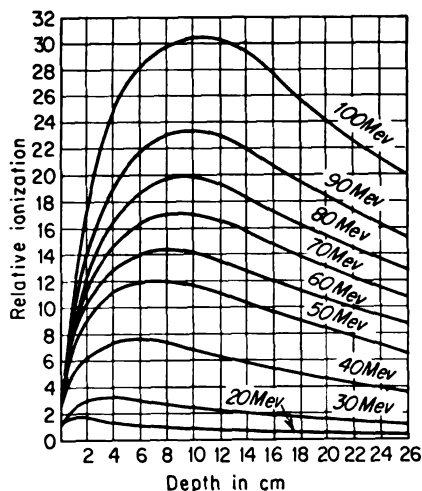


FIG. 13-3. The relative ionization (dosage) at any depth in a tissue-equivalent phantom produced by a unit entrance dose, at energy levels from 20 to 100 Mev. (By permission of Charles C Thomas, Publisher, Springfield, Ill. Charleton, E. E., and H. E. Breed, *Some Depth Dose Studies of Roentgen Rays for Energy Levels from 20 to 100 Million Electron Volts*, *Am. J. Roentgenol. Radium Therapy*, vol. 60, pp. 158-174, 1948.)

the minimum dose in the body (see depth dose curves in Fig. 13-3).

Unique **protection problems** arise as a result of the fact that the multimillion-volt radiation is capable of initiating nuclear reactions with the subsequent production of radioactive materials and neutrons, the latter of which require special shielding considerations (see section on Shielding). Methods of handling the unique problems associated with multimillion-volt therapy installations are covered extensively in the recommendations of the NCRP and set forth in NBS Handbook 55 and should be referred to by those interested in using this modality of treatment.

Barriers for multimillion-volt therapy installations may be as thick as 8 ft of concrete. Location of the therapy room as far from occupied areas as possible will effect considerable savings as will the use of earth shielding, such as achieved when the unit is located below ground level. A maze is generally used to cut down the amount of shielding required in the access door. Also, since these units do not

the evaluation of patient dosage by means other than the skin effect, the criterion which is used in ordinary deep therapy.

The **multimillion-volt therapy** range extends from 3 million volts to as high as 1 billion volts. However, the usual medical use of such radiation is in the range from 20 to 100 million volts. These energies are, at present, achieved mainly with betatrons (Harvey *et al.*, *Betatron Cancer Therapy*, *Radiology*, vol. 58, pp. 23-34, 1952) and with linear accelerators (Newberry, G. R., and D. K. Bewley, *Performance of Medical Research Council 8 Mev Linear Accelerator*, *Brit. J. Radiol.*, vol. 28, pp. 241-251, 1955). Although both these generators are capable of producing energetic charged-particle radiation, they are used in the medical field mainly as sources of X-rays. As is true in the supervoltage range, the scattered radiation is predominantly forward, and thus sharply defined beams and minimum side scatter are present. For this range, the maximum dose in the patient occurs at depths of from 0.5 to 10 cm. Starting at about 30 Mev, the entrance dose is

generally include a tube enclosure, it is recommended that for medical use a barrier be provided between the unit and the patient so as to cut down all leakage radiation to the patient to 0.1 per cent of the useful-beam value. The useful beam, of course, must be allowed to pass through an orifice in this barrier.

High-energy-particle Therapy Installations. The biological and therapeutic action of high-energy particles has been investigated in this country since 1936. The first studies concerned the biological action of neutrons (Lawrence, J. H., and E. O. Lawrence, Biological Action of Neutron Rays, *Proc. Natl. Acad. Sci. U.S.*, vol. 22, pp. 124-133, 1936) and the therapeutic use of neutrons in the treatment of malignant disease (Stone *et al.*, Preliminary Report on the Use of Fast Neutrons in Treatment of Malignant Disease, *Radiology*, vol. 35, pp. 322-327, 1940). Since these original studies, many uses of high-energy particles, both biological and therapeutic, have been investigated. A summary of some of these uses up to January, 1952, is included in the reference by Tobias *et al.* (Radiological Use of High Energy Deuterons and Alpha Particles, *Am. J. Roentgenol.*, vol. 67, 1952.) The use of high-energy particles in medical therapy may still be considered to be in the experimental stage, although several therapeutic studies have been carried out to date. The devices necessary for producing beams of these high-energy particles are large, complicated, and quite costly, and thus their use will necessarily be restricted to large centers such as the National Laboratories administered by the Atomic Energy Commission (AEC).

The **types of high-energy particles** which have been used therapeutically include neutrons, electrons, deuterons, alpha particles, and protons. The nuclear reactor or pile is an excellent source of neutrons, and as a result of this, reactors specially adapted to the treatment of patients have been devised and are in use. The major sources of electrons are the betatron and the linear accelerator (with special provision for removal of the electron beam from the accelerator). The major sources for deuterons, alpha particles, and protons are the cyclotron or synchrocyclotron and the linear accelerator.

The **biological effects of neutrons** result from their collisions with charged particles (mainly protons) in matter or from their ability to initiate nuclear reactions with the subsequent emission of energetic charged particles. In both instances, the biological effects are due predominantly to the interaction of the charged particles with matter. In a recent mode of neutron therapy, known as capture therapy (Farr *et al.*, Neutron Capture Therapy with Boron in the Treatment of Glioblastoma Multiforme, *Am. J. Roentgenol.*, vol. 71, pp. 279-293, 1954), this effect has been enhanced by the installation of boron-containing materials which increase the interaction of neutrons in the selected location, thus increasing the number of heavy charged particles (alpha particles in this case) available for interaction with matter.

The **biological effects due to high-energy charged particles** such as electrons, deuterons, alpha particles, and protons emitted by particle accelerators result from a special type of interaction of these particles with matter (see Sec. 7). Because of their high energy (from about 6 to 400 Mev), these particles lose energy rather slowly until their velocity decreases near the end of their range, and at that point their rate of energy loss becomes very high. This phenomenon results in the production of very high dosages near the ends of the ranges of the particles with very much smaller doses being produced at shorter distances or lesser depths.

The **protection procedures for high-energy-particle generators** are quite specialized and will be covered in a future report of Subcommittee 4 of the NCRP and also in NBS Handbook 55. In general, barriers, distance, and other usual protection devices are adapted to the special needs of these installations. Some of the special problems which occur are the production of neutrons by all the types of generators and the initiation of nuclear reactions by all the types of particles with resulting residual radioactivity and alteration of the radiations which are present. Some measurements made on various particle generators are included in the report of Solon *et al.* (Stray Radiation Measurements at Particle Accelerator Sites, *AEC Rept.* NYO-4699, June 1, 1956).

Rotational therapy (Hawley, S. J., *Rotation Therapy: Theory and Clinical Applications*, *Radiology*, vol. 51, pp. 205-213, 1948) is a mode of therapy in which either the X-ray beam is fixed and the patient is rotated about an axis in this beam, or the patient is fixed and the radiation source is caused to rotate about an axis in the patient. The axis, for the most part, is located in the center of the volume to which treatment is being administered. Rotational therapy is carried out mainly in the deep and supervoltage ranges, and with teletherapy units (see paragraph on teletherapy units). However, it may also be used to advantage in the multimillion-volt therapy range. The use of rotational therapy will allow the administration of greater doses to the volume being treated than will stationary-field techniques. However, this increased dose is achieved at the expense of a large volume of tissue which is treated to a somewhat lower degree. This large total energy absorption, or **integral dose** (Boag, J. W., *Physical, Biochemical and Therapeutic Aspects of Volume Dose*, *Brit. J. Radiol.*, vol. 18, pp. 240-246, 1945), as it is known, should be considered when rotational-therapy techniques are being contemplated, since it may seriously affect the over-all result of the treatment.

With the radiation beam fixed and the patient rotating within the beam, usual **protection procedures** can be followed. However, when the patient is kept stationary and the radiation source is rotated about the patient, it is obvious that the radiation beam may strike many more areas than in the case of a fixed beam. Primary-barrier protection, therefore, needs to be extended to cover all the areas which the useful beam may strike; this will include floor, ceiling, and walls. Many units (mainly teletherapy units) designed for moving-beam therapy include an integral shield located within the useful beam, but beyond the patient position. This integral shield is in the path of the useful beam for all its orientations during rotation, thus decreasing the amount of shielding that need be applied in the walls, ceiling, and floor of the room. If this latter type of rotational-therapy equipment is to be used, the advisability of installing an integral shield of this type should be investigated to determine whether it is economically advantageous, and whether its installation will limit the flexibility of the apparatus too greatly.

Patient-protection Devices in Therapy. A number of devices have been developed since the advent of X-ray therapy, all of which have had the purpose of assuring that the patient receive the proper dosage to the proper location with a minimum possibility of error. In order to assure treatment of the proper area, adequate **beam-defining devices** and **localizing devices** have been developed. The beam-defining devices include cones and diaphragms. The usual localizing device consists of a beam of light which is carefully adjusted so as to be coincident with the X-ray

beam, and which thus delineates the field being treated as a light area on the surface of the patient. In order to assure that the proper amount of dosage is delivered to the patient, the unit should be supplied with an accurate **timing device** which will cut off the X-ray beam after a preset period of time, which corresponds to a preset amount of dosage as determined from the calibration of the therapy apparatus. In addition to this, it is recommended that all deep-therapy units be supplied with a **radiation monitor** which will monitor the amount of dosage which has been delivered, or the rate at which the dosage is being delivered (see NBS Handbook 60). These monitors are generally of two types. The first is of the constancy type which is mounted within the X-ray tube head and calibrated for each treatment modality so as to indicate the emission of a constant amount of radiation. The second is the type which is placed upon the patient's skin, and which monitors directly the amount of radiation dosage which is delivered to the skin of the patient. In addition to timers and monitors, **filter indicators** are also of value in assuring that the proper dosage is delivered to the patient. These filter indicators provide an indication at the control panel, or other conspicuous location, that the proper filter is in place within the X-ray beam. All therapy equipment should be provided with a **view port** near the control panel which will allow the operator to view the patient throughout the treatment. This provides a means for ascertaining that the patient remains in the position that has been preset by the operator. It is also recommended that an oral device be provided so that the patient and the operator may communicate with each other during the treatment. The view port should contain protection equivalent to that within the wall or door in which it is included. Details for the installation of such view ports are included in the recommendations of the NCRP (NBS Handbook 60).

Warning Devices and Interlocks. Special devices are available for the protection of workers and other persons in the vicinity of X-ray therapy equipment. The NCRP (NBS Handbook 60) states that an interlock shall be provided at all access doors to X-ray deep-therapy rooms to cut off the radiation source when the doors to the room are open. In addition to this, a warning light may be provided over the doors to X-ray therapy rooms to indicate when the X-ray beam is on within the room; such lights are also valuable within the room to alert personnel who may be accidentally in the room while the unit is on, or who are in the room when the unit is accidentally turned on. It is also recommended by the NCRP that if any area outside a multimillion-volt installation is such that adequate protection is not afforded, warning lights and signs shall be installed to warn individuals that the location is one of radiation hazard during radiation procedures.

Calibrations. All X-ray therapy equipment should be subjected prior to use to a calibration performed by a qualified expert, as defined by the NCRP (NBS Handbook 60). This calibration should include a measurement of the exposure dose rate at all points in question, and for all anticipated settings of voltage, current, and filtration. It should also include a measurement of the quality (HVL) of the radiation beam, a determination of the uniformity of dosage rates within the X-ray beam, and an examination and adjustment of localizing and monitoring devices. A calibration chart signed by the qualified expert should be placed in a conspicuous location near the X-ray therapy controls. This procedure will provide assurance that proper radiation doses are administered and will also provide a valuable legal

document. It is recommended that the calibration procedure be repeated periodically and whenever major changes are made in the therapy equipment. Special calibration procedures for the multimillion-volt units are covered completely in NBS Handbook 55.

Surveys. All radiation-therapy installations should be subjected to a protection survey by a qualified expert prior to operation of the apparatus (see NBS Handbook 60 and 51). This survey should include an examination of the adequacy of the protective tube enclosure, of beam-defining devices, and of masking materials. It should also include a measurement of the radiation exposure at all positions around the X-ray installation. The survey and its associated measurements should be carried out in the manner prescribed by the NCRP. It is to be noted that special procedures are necessary for the survey of multimillion-volt therapy installations. These are covered in NBS Handbook 55. A report of the survey should be supplied to the institution requesting the survey, and to the person in charge of the X-ray installation. The report should include a compilation of the findings and measurements, as well as all recommendations as to barriers, procedures, equipment, personnel monitoring, etc. If the survey requires that changes be made, it will generally be necessary for a repeat survey to be conducted to assure that the changes have accomplished the required protection. It will be found that a great saving in time and expense will be achieved if a qualified expert is consulted to advise upon the planning of all radiation-therapy installations at the time of design.

RADIOACTIVE MATERIALS IN MEDICINE

Alpha-, beta-, and gamma-emitting radioactive materials are used in the medical field for various applications, as based upon their physical properties (see Secs. 5, 6, and 7 for more data on alpha, beta, and gamma rays).

Because of the very short range of alpha particles and their low penetration, **alpha emitters** cannot generally be used as external sources of radiation. As internal sources alpha particles produce very drastic biological effects as a result of their very rapid transfer of energy, and thus they are seldom used in this manner. Beta particles have a range in tissue up to a few centimeters and thus **beta emitters** can be used as external sources of radiation. In addition, they are also used extensively as internal emitters. The most commonly used radioactive materials in the medical field are the **gamma emitters**. Gamma rays are identical to X-rays except that they are emitted from radioactive nuclei in discrete energies or line spectra, rather than the continuous spectrum of energies emitted by X-ray tubes. Gamma rays, if used in the same manner as X-rays, may be expected to produce essentially the same biological results as produced by X-rays of equivalent energy.

The three types of radioactive materials listed in the paragraph above are used in a number of ways. The first of these is **teletherapy**—the use of large gamma-ray sources which are applied in the same manner as X-rays in the field of therapy. The second use is as a **surface source** and includes beta-ray applicators and various topical ointments and solutions containing radioactive materials. The third is the use of **interstitial sources**, generally gamma emitters, which are implanted as discrete sources in tissue at locations planned by the physician. These sources are generally enclosed in a metallic sheath which may be removed at a subsequent pre-

determined time. **Colloidal suspensions** of beta- and gamma-emitting isotopes (Au-198 and P-32) which are deposited in predetermined locations in the body may be included with interstitial sources, although they differ considerably in use from the classical interstitial sources. Fourth, and last, is the use of **internal sources** in which radioactive materials are introduced into the body and then are allowed to travel to locations determined by metabolic pathways in the body. This last use of isotopes differs from interstitial use in that, as soon as the introduction into the body is accomplished, the physician no longer has control over the deposition or elimination of the radioactive material, since this must then be carried out by normal body processes.

Teletherapy Installations. The use of high-intensity gamma-ray sources as substitutes for conventional X-ray sources in the field of therapy was started in the early 1920s (Lysholm, E., *Early Teletherapy Units, Acta Radiol.*, vol. 2, p. 516, 1923) and has become known as teletherapy. The first teletherapy units used radium in equilibrium with its decay products (see Sec. 6 for details on radium) in the form of a radium pack, and contained anywhere from a few to tens of curies housed in a shielded head provided with an orifice for the efflux of the gamma radiation. The nuclear reactor or pile has made available many artificially produced isotopes (for example, Co-60 and Cs-137) which lend themselves to teletherapy use and which are much less expensive than radium. The gamma radiations of radium, cobalt, and cesium range in energy from about 600 to 2,300 kev and are equivalent to the average energies of 1- to 3-million-volt X-ray beams. Teletherapy units using these three isotopes are used as and possess therapeutic properties similar to conventional supervoltage therapy units. A number of other artificially produced isotopes have been proposed as teletherapy sources and are listed in Table 13-1, along with some of their pertinent physical properties. Teletherapy units are

Table 13-1. Gamma-emitting Isotopes for Medical Use

Isotope	Half-life	Gamma-ray energies, Mev	Uses
Ra (+ daughters)	1,600 years	0.284-2.3	Teletherapy and interstitial
Cs ¹³⁷	33 years	0.662	Teletherapy
Co ⁶⁰	5.3 years	1.17 1.33	Teletherapy and interstitial
Eu ¹⁵² -Eu ¹⁵⁴	5.3 years	1.2	Teletherapy
Ir ¹⁹²	70 days	0.137-0.651	Teletherapy and interstitial
Au ¹⁹⁸	2.7 days	0.412	Interstitial

used in fixed-beam therapy and also lend themselves to rotational-therapy techniques (see paragraph on Rotational Therapy above) because of the minimum of electrical connections which must be made to the rotating radiation source. Special teletherapy units using multiple small sources with their radiation beams focused at a point have also been devised (Quick, D., and J. D. Richmond, *Preliminary Experiences with a 50 gm. Converging Beam Radium Unit, Am. J. Roentgenol.*, vol. 74, pp. 635-650, 1955) to accomplish the same type of results as are achieved with

rotational therapy. The NCRP has put forth a series of recommendations pertaining to the design, shielding, and the use of teletherapy units in NBS Handbook 54. These recommendations should be consulted for more details regarding teletherapy protection and safe usage. Radiation protection for teletherapy units is accomplished in much the same manner as for conventional therapy units; however, when artificially produced isotopes are used in teletherapy units, requirements of the AEC must also be met, since this agency licenses users for the purpose of obtaining these government-controlled isotopes.

Beta-ray Applicators. Many radioisotopes, both naturally occurring and artificially produced, have been used in beta-ray applicators (Krohmer, J. S., *Physical Measurements on Various Beta Ray Applicators*, *Am. J. Roentgenol.*, vol. 66, pp. 791-796, 1951). The beta-ray applicator contains a relatively large amount (10 to 100 mc) of beta-emitting isotope shielded in all directions except one by a relatively thick container. The one lightly shielded surface, which is known as the treatment face of the applicator, includes a very thin shield sufficient to enclose the isotope, but thin enough to allow the efflux of beta radiation. The beta rays which are emitted through this thin treatment face are used in the treatment of superficial conditions of the skin, and more commonly, of the eye, by placing the treatment face in direct contact with the area to be treated (Friedell *et al.*, *Beta Ray Application to the Eye, with Description of Applicator Utilizing Sr⁹⁰ and Its Clinical Use*, *Am. J. Ophthalmol.*, vol. 33, pp. 525-533, 1950). Because of the very short treatment distance, radiation dose rates are extremely high and treatment times are short. These short exposures minimize the radiation hazard to the operator and to the patient, provided the source is carefully used and is kept in a shielded container when not in use. Many of the sources used in these applicators also emit the more penetrating gamma radiation and thus make shielding problems greater and add to the exposure of operator and patient during use of the applicator. Even applicators containing pure beta emitters, such as Sr-90 and P-32, present some penetrating radiation hazard in the form of bremsstrahlung which are identical to gamma rays and which are produced by the interaction of beta particles with surrounding matter (see Sec. 7 on Interaction of Radiation with matter). The production of bremsstrahlung is quite inefficient in low-atomic-number materials, and thus these materials should be used for the applicator capsule and for the shielding used around the applicator. Ideal materials for this encapsulation and shielding are aluminum, wood, and plastics.

Table 13.2 lists beta-emitting isotopes which may be used for beta-ray applicators, along with a number of their pertinent physical properties. It should be noted that the majority of these isotopes have relatively long half-lives and thus lend themselves to encapsulation in essentially permanent sources. The one short-half-life material in the table (P-32) has been used in a number of ways as a short-lived beta-ray applicator. It has been used in solution applied through the medium of wet blotting paper and included in various plastics for the treatment of skin conditions (Low Beer, B. V. A., "The Clinical Use of Radioisotopes," Charles C Thomas, Springfield, Ill., 1950). As can be seen, the wet-blotting-paper technique presents additional hazards due to the possibility of spillage and contamination with the P-32 in solution. The sealed permanent sources present no such hazard and thus are to be preferred. However, they must be tested periodically to assure that no leakage of

isotope is present, since leakage of the long-lived isotopes would present a contamination problem of a very serious nature.

Table 13-2. Beta-emitting Isotopes for Medical Use

Isotope	Half-life	Important peak beta-ray energy, Mev	Gamma rays present
Ra C (Ra)	1,600 years	3.15	Yes
Ra C (Rn)	3.825 days	3.15	Yes
RaD + E	22 years	1.17	Yes
Sr ⁹⁰ -Y ⁹⁰	19.9 years	2.18	No
Ru ¹⁰⁶ -Rh ¹⁰⁶	1.0 year	3.55	Yes
Ce ¹⁴⁴ -Pr ¹⁴⁴	275 days	3.07	Yes
P ³²	14.3 days	1.712	No

Alpha Sources. One of the very rare uses of alpha radiation in medical therapy lies in the field of dermatology. In the past, and in recent years, very superficial skin conditions have been treated by the application of radon ointment (Delario, A. J., "Roentgen, Radium and Radioisotope Therapy," Lea & Febiger, Philadelphia, 1953) or a Thorium X solution (Witten *et al.*, The Mode of Action of Thorium X on Human Skin, *Am. J. Roentgenol.*, vol. 74, pp. 90-97, 1955). Both radon and Thorium X emit alpha particles which, because of their large mass and subsequent low velocity, lose energy very rapidly, and thus have an extremely short range (up to approximately 50 microns in tissue). These two alpha-ray sources are applied directly to the skin and deliver their treatment to the very superficial layers of the skin. Some of the materials, however, are absorbed into the skin and thus produce their characteristic biological effects at points well below the surface. The hazards from these materials lie principally in the possibility of contamination with subsequent possible ingestion by human beings. Both materials have very short half-lives (Rn, 3.83 days; Th-X, 3.65 days) and thus the associated hazards are also short in duration.

Interstitial Sources. In this mode of therapy, encapsulated or solid sources are implanted in predetermined tissue locations at the will of the physician. Although interstitial therapy pertains specifically to radiation sources implanted into the interstices of the tissue, we may also consider under this heading sources which are introduced into body cavities, or so-called intracavity therapy. In either of these modes of therapy, the radiation sources are implanted in predetermined patterns which will achieve a calculable dosage distribution within the tissue to be treated. Several systems of implantation are in use and are described completely in other literature (Glasser *et al.*, "Physical Foundations of Radiology," Hoeber, New York, 1952). Interstitial and intracavity therapy was accomplished through the use of radium or radon until recent years. With the availability of artificially produced radioisotopes, Co-60 has come into rather widespread use for both these types of therapy (Meschan *et al.*, Dosage Tables for Co⁶⁰ Use with Interstitial Plaque and Intracavitary Applications, *Am. J. Roentgenol.*, vol. 71, pp. 320-332.

1954). Other isotopes such as Iridium-192 have also been used (Henschke, U. K., *Proc. Intern. Conf. Peaceful Uses Atomic Energy, Geneva*, vol. 10, 1955). The physical types of applicators which are used in interstitial therapy are needles, seeds, tubes, capsules, wires, and specially loaded nylon threads or tubes.

The principal hazards to personnel in **interstitial therapy** occur at the time of **source preparation**, and at the time of implantation. Since the patterns to be achieved are at times quite intricate, the implantation procedure may be quite time-consuming and present considerable hazard to both physician and his aides. It may often be necessary to carry out an implantation procedure in shifts in order that no one person will be exposed to an excessive radiation dose. Handling of the sources should always be carried out with forceps and other distance-providing devices, and properly shielded containers should be provided for use at the time of implantation in order to cut down the amount of exposure. Properly shielded source-preparation areas should also be provided.

Interstitial and Intracavity Therapy with Colloidal Suspensions. Radioactive sources, such as colloidal gold (Au-198) and colloidal chromic phosphate (P-32) are used for the treatment of malignant disease as both interstitial and intracavity sources (Hahn, P. F., "Therapeutic Use of Artificial Radioisotopes," Wiley, New York, 1956). In the case of interstitial use of colloids, the colloid is introduced by hypodermic injection into the tissue to be treated. As intracavity sources the colloids are introduced into the body cavity (e.g., the abdominal cavity or the thoracic cavity) through a previously inserted trocar. In either the interstitial or intracavity type of use, these colloidal radioactive substances may be considered to be most similar to interstitial radiation sources as covered above, since they are introduced into locations predetermined by the physician. In this case, however, the sources cannot be removed as can ordinary interstitial sources but must instead remain in place, until the radioactive material has disintegrated completely. As a result of this, considerable care must be exercised in the introduction of these sources into the patient in order that they be instilled into the proper location. Careful use of these materials must also be carried out to avoid the possibility of contamination. Special syringes and instillation devices must also be available since the activities which are used vary from 2 to 100 mc and thus represent a definite hazard to personnel. The possibility of leakage of radioactive colloid from the patient must also be avoided in order to eliminate contamination.

Internal Sources. The use of internal radiation sources in the fields of medical diagnosis and therapy concerns sources which are administered to the patient, either by mouth or by injection, and which from that time on are distributed within and excreted from the body by normal metabolic processes. This type of isotope use is relatively new, the first sources becoming available through the cyclotron production of such isotopes. At present, most of the isotopes used in this manner are produced in the nuclear reactor or pile through the medium of neutron bombardment (see Sec. 6).

Although many isotopes have been used for many diagnostic and therapeutic procedures in the last 10 years, relatively few have found a permanent place in the fields of proved medical diagnosis and therapy. **Radioactive Iodine-131** is by far the most important of the isotopes being used in the medical field as internal emitters. Its uses include the evaluation of thyroid function, treatment of hyper-

thyroidism, treatment of various heart disorders through obliteration of the thyroid gland, measurement of cardiac output, determination of flow rates, determination of adequacy of peripheral blood flow, measurement of blood volume, scanning or visual outlining of various body organs, and other similar studies. In these many uses, the radioactive iodine is either used as an inorganic iodide or combined into some organic compound such as human serum albumin or rose bengal dye. The next most important medically used isotope is **Phosphorus-32**, which is used in the treatment of various blood conditions such as polycythemia and leukemia, treatment of certain forms of cancer, and in the identification and localization of malignant tumors. A discussion of many of the internal uses of these two radioisotopes, and many others which have been used for medical diagnosis and therapy, is included in the texts by Low Beer (Low Beer, B. V. A., "The Clinical Use of Radioactive Isotopes," Charles C Thomas, Springfield, Ill., 1950) and Hahn (Hahn, P. F., "Therapeutic Use of Artificial Radioisotopes," Wiley, New York, 1956), and in the usual medical literature. These sources should be consulted for more details.

The **principal hazards of internal radioisotopes** in therapy lie in (1) possible overexposure of the patient; (2) contamination of areas in which isotopes are used, handled, and stored; and (3) the possible overexposure of personnel during preparation for, and carrying out of, the particular procedure. Complete dosage calculations should always be carried out prior to using an internal emitter in patients to determine whether the incidental exposure to the patient will be hazardous. In evaluating the degree of hazard to the patient, consideration must be given to the value of the test and the results which may be achieved by the test. It may often be true that the results to be received from the test are so important as to warrant a calculated overexposure of the patient. Accurate assay methods and dose preparation must then be followed to assure that the patient receives the proper dose as based upon calculations. Care must be exercised in the preparation and administration of the isotope to avoid the possibility of personnel and area contamination. Adequately shielded storage and handling facilities should be provided, and all personnel should be thoroughly acquainted with the possible hazards associated with the use of these radiation sources.

Isotopes Committee. Since most of the artificially produced radioisotopes are produced in government-controlled nuclear reactors, the AEC supervises the distribution and licensing procedures for these materials. In order for an institution to become licensed for receiving and using radioactive materials, the AEC requires that the institution establish a Radioisotopes Committee within the institution (Atomic Energy Commission, Rules and Regulations: Licensing of Byproduct Materials, *Federal Register*, vol. 21, pp. 213-217, Jan. 11, 1956). This committee is responsible to the AEC for guaranteeing the safe handling and proper usage of radioisotopes within the institution. The Radioisotopes Committee is to accomplish this purpose by reviewing all requests for use of isotopes, by advising in their proper use, and by assuring themselves that all users are properly informed as to the hazards, possibilities, and limitations associated with the use of radioactive materials. The AEC recommends that the Committee include a therapeutic radiologist, an internist, a hematologist, and a person experienced in assay of radioisotopes and protection against ionizing radiation. They also recommend that other fields of specialization be represented on the committee, depending upon the programs to be

carried out. It has also been recommended elsewhere (Low Beer, B. V. A., *Organizing and Controlling the Use and Safe Handling of Isotopes, Hospitals, J. Am. Hospital Assoc.*, December, 1955) that the Committee include a supervisor of nursing and a member from the administrative branch of the hospital. Inclusion of the supervisor of nursing will aid in the carrying out of the radioisotope program, since the nurses will then feel more a part of the program and will be better informed as to its scope and requirements. The hospital administrator will be able to provide liaison with the board of directors and will be able to provide this body with first-hand information which they may require in the formulation of policies. Even in the event that isotopes are not procured from government-controlled sources, it is recommended that such a committee be provided in all institutions, since its existence will guarantee proper and safe usage of radioactive materials. In regard to membership of the committee, it should be stated that the recommendations are quite flexible and may be modified in accordance with the availability of the personnel at the institution. (Note: For requirements concerning noninstitutional or private use of isotopes, see AEC regulations cited above.)

Dose to Patient. It is of utmost importance that all measurements, calculations, and controls of patient dosage be carefully carried out so that optimum results of diagnostic and therapeutic procedures can be achieved without unnecessary and potentially hazardous radiation effects. This is especially true in diagnostic procedures where the radiation is used simply as the *modus operandi* for obtaining the results, and has no beneficial effect whatsoever. In this case, the unavoidable radiation dosage should be kept to as low a value as possible without sacrifice to the accuracy of the procedure being carried out. The radiation dosage and the possible effects of this dose should be thoroughly evaluated, and the findings should be included in all decisions concerning the advisability of carrying out the test. In the case of therapy, the radiation itself is being used as the means of treatment, and protection of the patient is carried out mainly by limiting the amount of tissue which is exposed to the effects of the radiation, and by carefully controlling or calculating the dosage to which the patient will be exposed. In the case of **external sources**, patient protection is best afforded by careful calibration of the radiation source and careful control of radiation factors such as distance, field size, and time of treatment. Evaluation of dosage distribution with external sources is made possible by the use of properly prepared isodose curves which are available for the more commonly used external sources. **Interstitial and intracavity sources** are invariably used for therapeutic purposes, and therefore the patient-protection problem is one of proper evaluation of the level and distribution of radiation dose. In this case, calculation of dosage and implantation of sources should best follow a properly evaluated and accepted system of interstitial therapy. Isodose curves and dosage-distribution data are available for all the accepted systems. More detailed discussions of these systems appear in the following reference (Glasser *et al.*, "Physical Foundations of Radiology," Hoeber, New York, 1952). Interstitial and intracavity sources (except for colloidal suspensions) must be removed at a predetermined time, dependent upon dosage calculations, if proper dose is to be delivered. Dosage calculations for **interstitial and intracavity use of colloidal suspensions** are by no means well developed, but estimates of dosage are included in literature on the subject. The physician's control over an **internal emitter** ceases immediately upon administration

of the isotope to the patient, and thus all precautionary measures must be carried out in advance of this procedure. Careful calculation of the possible dosage to the patient should be carried out and should include evaluation of dosage to all parts of the patient, even though the diagnostic or therapeutic procedure may be limited to one particular organ or location. Since the internal emitter may possibly go to any portion of the body, this calculation requires an accurate knowledge of the type of emission, the energy of the emission, the biological half-life of the isotope or labeled compound being used, and the distribution of that isotope within the body. The last two quantities are the most difficult to evaluate, and in general, data on biological half-life and distribution are not available and must therefore either be determined or reasonable approximations made thereof. Presently accepted methods of dosage calculation for internal isotopes are discussed fully in the textbook by Glasser *et al.* and in other sources.

Patient Care. Special procedures in the care of patients covered in this paragraph pertain mainly to patients who have been given, or have had implanted, therapeutic quantities of radioactive materials, and not to those who have received diagnostic or tracer amounts of radioactive material. Patients undergoing diagnostic or tracer studies receive very small quantities of isotopes and thus require no special care and are generally treated on an outpatient basis. Even if the diagnostic study is carried out on inpatients, these patients may receive only normal patient care, with one exception. This occurs when the diagnostic inpatient must use a bedpan, for then special care must be exercised to avoid radioactive contamination (see below). For patients undergoing **interstitial or intracavity therapy**, the main hazard lies in the possibility of exposure of nurses or other personnel to harmful amounts of radiation due to the generally large amount of radioactive material within the patient. The area around these patients should be monitored, and the nursing staff should be made aware of the radiation levels which exist. The care which these patients receive should then be dependent upon the radiation levels and should be provided in such a way that nurses and other personnel are not exposed to radiation above the maximum permissible level. This may be accomplished by the use of special shields, by distance protection, and by limiting the time during which any one person can be in a given vicinity. Interstitial and intracavity sources are generally removed in the patient's room, and thus there should be provided adequate forceps and other distance-providing devices as well as an adequately shielded container for receiving the radioactive sources. The care of patients containing **colloidal suspensions of radioactive materials** is essentially the same as for interstitial and intracavity sources, except for the possible leakage of activity. In this respect, these sources present problems similar to those discussed below for internal emitters. When therapeutic quantities of **internal emitters** are used, the problems in patient care are twofold. First, the hazard from radiation is similar to that associated with the interstitial and intracavity sources, and in addition, the problems of disposal of excreta and contamination of bed linens, etc., are also present. In this case, the excreta will contain large quantities of radioactive material and must be very carefully handled in order to avoid contamination and radiation exposure. The general procedure will be for the person handling bedpans, etc., to wear rubber gloves to avoid self-contamination and to exercise rapid but careful disposal of this excreta. If the excreta is not to be saved for further measurement, it may generally be dis-

posed of in existing toilets, provided the institution involved is a large one with a large flow of water through existing sewers (see NBS Handbook 49 and Code of Federal Regulations, title 10, pt. 20). If the material is to be saved, adequately shielded containers should be provided for storing until measurements can be carried out. When large quantities of radioactive materials are disposed of down existing drains, this should be accomplished with the flow of copious quantities of water and may periodically be followed by the addition of suitable amount of **carrier**, that is, nonradioactive isotope of the element which has been used. The water flush provides for usual dilution of the radioactive material, while the addition of carrier isotope provides for isotopic dilution. In order to avoid excessive contamination of bedding, mattresses, etc., it is recommended that rubber sheeting be used below sheets and in all other areas in the room in which possible contamination might occur. Bedding which has been in contact with a patient containing therapeutic quantities of radioactive materials should be set aside and should be surveyed prior to being put through normal laundry channels. If the bedding is found to be contaminated, it should be held until radioactive decay brings the level of contamination down to permissible amounts. Every institution in which therapeutic quantities of radioactive materials are used, and in which patients containing these amounts are hospitalized, should have a set of rules prepared for nursing care ("Procedures to Be Carried Out When Radioisotopes Are Administered to Patients," National Institute of Health, Radiation Safety Committee, Feb. 23, 1954). These rules should cover all phases of patient care and should acquaint the nurse with the problems associated with each type of isotope use. In addition, there should be appended to these rules emergency plans for the various emergencies which can possibly arise. All persons participating in the care of patients containing radioactive materials should be required to read the rules and should be properly informed as to the properties, uses, and safe handling of radioactive materials.

Safe storage and handling facilities (see also Secs. 23 and 25, and NBS Handbook 42) shall be provided at all locations where radioactive materials are to be handled or stored. These facilities shall make proper use of distance and shielding to provide safe and efficient storage and handling procedures. Storage containers for gamma-emitting isotopes will generally be constructed of lead, concrete, or steel, while those for beta and alpha emitters will generally be composed of low-atomic-number materials, such as plastic, wood, and water. Adequate storage containers for most isotope uses are commercially available or easily fabricated. Safe-handling facilities may consist only of a simple distance-providing device such as a pair of tongs or forceps. However, where source preparation is involved, a more elaborate system may have to be provided. This may consist of a shielded glove box, or possibly a shielded work bench, equipped with a mirror or leaded-glass window for visual control of procedures being carried out. These devices may also be procured commercially, either in a ready-made form, or to the customer's designs and specifications. A shielded cart or carrier is also required to transport radioactive materials from one location to another.

Control of radioactive contamination in the medical field, as well as in other types of uses, is very important, both from the standpoint of protection of personnel and also for the purpose of maintaining a low background level which will not affect the accuracy and ease of carrying out the procedure. The greatest hazard to the worker

in the field of radioactivity lies in the accidental entry of radioactive material into the worker's body. The mode of entry may be inhalation, ingestion, accidental injection, or entry through an open wound on the body. Regardless of the mode of entry, a small amount of internally deposited isotope may present a considerable hazard, whereas the same amount would be unobjectionable if outside the body. It is because of this fact that great care must be exercised in carrying out all radioactive procedures to be sure that no contamination occurs. This may be accomplished by careful planning of procedures, by use of proper equipment, and by the use of proper survey procedures, both during and after the procedure has been carried out. When contamination has been found to occur, an immediate effort should be expended to effect decontamination (see Sec. 18). If this is not possible, the material or area must be set aside or marked off so that the contamination is not allowed to spread. If the nonremovable contamination is due to a short-lived isotope, simple passage of time will correct the situation. If it is due to a long-lived isotope, extensive measures may be necessary to overcome the problem. It is advisable, because of the foregoing situation, to use materials which are easily decontaminable, if possible. Asphalt, rubber, or vinyl-tiled floors are ideal, since decontamination may be effected by simply replacing tiles. Stainless-steel and other nonporous sinks, surfaces, and utensils are also easily decontaminated, and thus are recommended for use with radioactive materials. Whenever possible, porous materials such as concrete, wood, and cloth should be avoided because they are the most difficult to decontaminate. Materials, precautions, and procedures necessary in the control and removal of contamination are covered quite completely in NBS Handbook 48 and in Sec. 18. These sources should be consulted for more complete details.

Waste Disposal. The subject of radioactive-waste disposal is covered completely in Sec. 21 of this handbook. However, it was deemed advisable to include special considerations for the medical use of isotopes in this section. The usual problem is the disposal of excreta from patients who have been given therapeutic quantities of isotopes. The isotopes generally encountered are P-32 and I-131, the disposal of which is covered completely by the NCRP in NBS Handbook 49. These materials and other liquid wastes are best disposed of by flushing down existing drains, including toilets. Adequate dilution is assumed to be obtained in this manner, but recommendations of the NCRP should be consulted prior to embarking upon a radiation program. Other disposal problems which may come up are the disposal of contaminated materials, both burnable and nonburnable. These materials should, of course, be monitored prior to any disposal in order to determine the levels which are to be handled. It will generally be found that burnable materials such as gauze bandages, cotton, and other cloth materials which have been contaminated, can be incinerated by usual methods. The solid nonburnable materials will generally require disposal by more elaborate means, such as burial or disposal at sea. Recommendations for the latter types of disposal are covered in NBS Handbook 58 and in recommendations of the AEC ("Handling Radioactive Wastes in the Atomic Energy Program," AEC, Washington, D.C., 1951).

Radiation Surveys. Proper radiation survey methods, as recommended by the NCRP in NBS Handbook 51, will aid greatly in the control of contamination and exposure to personnel as encountered in the medical field. General recommenda-

tions are that all areas and locations where radioactive materials are handled, transported, stored, or used should be monitored periodically in order to evaluate hazard to personnel through radiation exposure and through contamination. All medical radiation procedures should be monitored throughout the procedure, and all equipment and areas checked thoroughly after the procedure and during the subsequent hospitalization of the patient. This will include the monitoring of rooms, beds, bedding, bedpans, toilets, waste containers, sinks, etc. The importance of proper surveying and monitoring procedures cannot be overemphasized in the management of a good medical radioisotopes program.

General References

- Glasser, O., E. H. Quimby, L. S. Taylor, and J. L. Weatherwax, "Physical Foundations of Radiology," Hoeber, New York, 1952.
- Hahn, P. F., "The Therapeutic Use of Artificial Radioisotopes," Wiley, New York, 1956.
- Hine, G. J., and G. L. Brownell, "Radiation Dosimetry," Academic Press, New York, 1956.
- Johns, H. E., "The Physics of Radiation Therapy," Charles C Thomas, Springfield, Ill., 1953.
- Low Beer, B. V. A., "The Clinical Use of Radioactive Isotopes," Charles C Thomas, Springfield, Ill., 1950.
- Morgan, R. H., and K. E. Corrigan, "Handbook of Radiology," Year Book Publishers, Chicago, 1955.
- Proc. Intern. Conf. Peaceful Uses Atomic Energy, Geneva*, vol. 10, 1956.
- Scott, W. G., "Planning Guide for Radiologic Installations," Year Book Publishers, Chicago, 1953.
- Wilson, C. W., "Radium Therapy," Baillière, London, 1956.
- Ferlazzo *et al.*, Radiation Hazard Evaluation and Control in Hospitals, *Radiology*, vol. 65, pp. 892-902, 1955.
- Wilsey, R. B., The Use of Photographic Films for Monitoring Stray X-rays and Gamma Rays, *Radiology*, vol. 56, p. 229, 1950.
- Looney, W. B., Late Clinical Changes Following the Internal Deposition of Radioactive Materials, *Ann. Internal Med.*, vol. 42, pp. 378-387, 1955.
- Morgan, R. H., Screen Intensification: A Review of Past and Present Research with an Analysis of Future Development, *Am. J. Roentgenol.*, vol. 75, pp. 69-76, 1956.
- Hirsch, I. S., Fluorography, the Photography of the Fluoroscopic Image, *Am. J. Roentgenol.*, vol. 43, p. 45, 1940.
- Editorial, Shoe Fitting Fluoroscopes, *J. Am. Med. Assoc.*, vol. 15, pp. 1004-1005, 1949.
- Moyers *et al.*, Radiation Exposure in Shoe Stores, *N.Y. State Dept. Labor, Monthly Rev., Div. Ind. Hyg.*, vol. 31, pp. 33-40, 1952.
- Safety Code for the Industrial Use of X-rays, American Standards Association, 254:1, 1946.
- Safe Operation of X-ray Shoe Fitting Machines, *Bull. Bur. Ind. Hyg., Dept. Public Health Welfare*, Cleveland, Ohio.
- Hempelmann, L. H., Potential Dangers of the Uncontrolled Use of Shoe-fitting Fluoroscopes, *New Engl. J. Med.*, vol. 241, p. 333, 1949.
- Laughlin, J. S., and I. Pullman, Physical Properties of Ionizing Radiations Which Affect the Population of the U.S., Preliminary and Unclassified Report to the Genetics Panel of the National Academy of Sciences' Study of the Biological Effects of Atomic Radiations, Apr. 18, 1956.

- Glasser, O., The Physical Foundation of Grenz-ray Therapy, *Radiology*, vol. 18, p. 713, 1932.
- Howes, W. E., and M. R. Camiel, Contact Roentgen Therapy, *Am. J. Roentgenol.*, vol. 48, p. 360, 1942.
- Cipollaro, A. C., Beryllium Window Radiations for Superficial Therapy, *Arch. Dermatol. and Syphilol.*, vol. 62, pp. 214-221, 1950.
- Harvey *et al.*, Betatron Cancer Therapy, *Radiology*, vol. 58, pp. 23-34, 1952.
- Newbery, G. R., and D. K. Bewley, Performance of Medical Research Council 8 Mev Linear Accelerators, *Brit. J. Radiol.*, vol. 28, pp. 241-251, 1955.
- Lawrence, J. H., and E. O. Lawrence, Biological Action of Neutron Rays, *Proc. Natl. Acad. Sci. U.S.*, vol. 22, pp. 124-133, 1936.
- Stone *et al.*, Preliminary Report on the Use of Fast Neutrons in Treatment of Malignant Disease, *Radiology*, vol. 35, pp. 322-327, 1940.
- Tobias *et al.*, Radiological Use of High Energy Deuterons and Alpha Particles, *Am. J. Roentgenol.*, vol. 67, 1952.
- Farr *et al.*, Neutron Capture Therapy with Boron in the Treatment of Glioblastoma Multiforme, *Am. J. Roentgenol.*, vol. 71, pp. 279-293, 1954.
- Solon *et al.*, Stray Radiation Measurements at Particle Accelerator Sites, *AEC Rept. NYO-4699*, June 1, 1956.
- Hawley, S. J., Rotation Therapy: Theory and Clinical Applications, *Radiology*, vol. 51, pp. 205-213, 1948.
- Boag, J. W., Physical, Biochemical and Therapeutic Aspects of Volume Dose, *Brit. J. Radiol.*, vol. 18, pp. 240-246, 1945.
- Lysholm, E., Early Teletherapy Units, *Acta Radiol.*, vol. 2, p. 516, 1923.
- Quick, D., and J. D. Richmond, Preliminary Experiences with a 50 gm. Converging Beam Radium Unit, *Am. J. Roentgenol.*, vol. 74, pp. 635-650, 1955.
- Krohmer, J. S., Physical Measurements on Various Beta Ray Applicators, *Am. J. Roentgenol.*, vol. 66, pp. 791-796, 1951.
- Friedell *et al.*, Beta Ray Application to the Eye, with Description of Applicator Utilizing Sr^{90} and Its Clinical Use, *Am. J. Ophthalmol.*, vol. 33, pp. 525-533, 1950.
- Delario, A. J., "Roentgen, Radium and Radioisotope Therapy," Lea & Febiger, Philadelphia, 1953.
- Witten *et al.*, The Mode of Action of Thorium X on Human Skin, *Am. J. Roentgenol.*, vol. 74, pp. 90-97, 1955.
- Meschan *et al.*, Dosage Tables for Co^{60} Use with Interstitial Plaque and Intracavitary Applications, *Am. J. Roentgenol.*, vol. 71, pp. 320-332, 1954.
- Henachke, U. K., *Proc. Intern. Conf. Peaceful Uses Atomic Energy, Geneva*, vol. 10, 1955.
- Atomic Energy Commission, Rules and Regulations: Licensing of Byproduct Materials, *Federal Register*, vol. 21, pp. 213-317, Jan. 11, 1956.
- Low Beer, B. V. A., Organizing and Controlling the Use and Safe Handling of Isotopes, *Hospitals, J. Am. Hospital Assoc.*, December, 1955.
- Procedures to Be Carried Out When Radioisotopes Are Administered to Patients, National Institute of Health, Radiation Safety Committee, Feb. 23, 1954.
- Laughlin, J. S., M. L. Meurk, I. Pullman, and R. S. Sherman, Bone, Skin and Gonadal Doses in Roentgen Diagnostic Procedures, *Am. J. Roentgenol.*, vol. 78, p. 961, 1957.
- Fields, T., and L. Seed, "Clinical Use of Radioisotopes," Year Book Publishers, Inc., Chicago, 1957.
- Beierwaltes, W. H., P. C. Johnson, and A. J. Solari, "Clinical Use of Radioisotopes," W. B. Saunders Company, Philadelphia, 1957.
- Quimby, E. H., S. Feitelberg, and S. Silver, "Radioactive Isotopes in Clinical Practice," Lea & Febiger, Philadelphia, 1958.

Section 14

DETERMINATION OF EXPOSURES

By

KARL Z. MORGAN, Ph.D., *Director, Health Physics Division, Oak Ridge National
Laboratory, Oak Ridge, Tennessee.*

Radiation from External Sources.....	14-2
Exposure Dose.....	14-2
Absorbed Dose from Internal Sources.....	14-10
Comments on the Critical-organ Concept.....	14-18

DETERMINATION OF EXPOSURES

Karl Z. Morgan

RADIATION FROM EXTERNAL SOURCES

See page 14-18 for General References.

There are two types of "dose" used in radiation hygiene, i.e., **exposure dose** and **absorbed dose**. The term exposure dose applies only to the measurement in roentgens (r) of X- and gamma-radiation intensity, and the absorbed dose applies to the measurement of energy delivered per gram of the medium by any type of ionizing radiation. The absorbed dose was originally measured in reps, 1 rep being an absorbed dose of 93 ergs/g of soft tissue (see also Sec. 2), but the rep unit has been replaced generally by the **rad**, 1 rad being an absorbed dose of 100 ergs/g of any medium. For convenience in keeping personnel-exposure records, the absorbed dose is frequently recorded in **rem** units [number of rem = (RBE) $\times$ number of rad]. Although this rem unit is extremely useful for personnel monitoring, it should not be used generally for recording experimental biological data because one must know the number of rads delivered by each type of radiation in order to describe adequately the radiation hazard.

The exposure dose or absorbed dose delivered by any ionizing radiation can be reduced by reducing the *time* of exposure, by increasing the *distance* from the source, or by increasing the *attenuating material* between the source and the point of measurement.

EXPOSURE DOSE

X- and gamma radiation are electromagnetic- (or wave-) type radiations, and the **exposure dose (r) from a point source** that emits monochromatic (single-wavelength) X- or gamma radiation is given by the equation

$$D_p = \frac{3.6(\mu - \sigma_a)_a E_1 e^{-\mu_1 y} b_1 B_1 t 10^9}{X^2} = \frac{R_1 e^{-\mu_1 y} B_1 t}{X^2} \quad (14-1)$$

in which D_p is the exposure dose in roentgens (r) delivered by a 1-curie point source at a distance X (cm). This monochromatic source emits b_1 photons of energy E_1 (Mev) per disintegration, and these photons are attenuated by the total absorption coefficient μ_1 (cm⁻¹) in the shielding material of thickness y (cm). The term $(\mu - \sigma_a)_a$ is the total coefficient of absorption (cm⁻¹) in air minus the Compton-

scattering coefficient of absorption (cm^{-1}) in air, B_1 is the build-up factor, R_1 is the r/day at 1 cm distance from the 1-curie point source, and t is the time in days the dose is delivered. In this section the day is taken as a 24-hr period rather than an 8-hr workday. If there is no shielding material other than air and if X is small,

$$e^{-\mu_1 X} B_1 \approx 1 \quad (14-2)$$

This approximation is correct within 5 per cent if the distance < 200 cm for radiation of energy $E_1 > 100$ kev. The value of B_1 is a function of the width of the beam, distance of the source and volume, atomic number, and position of the scattering medium. Figure 14-1 illustrates how the build-up varies with the energy of the radiation and with depth in a phantom. In this case, the ordinate of the curves is the per cent of the surface exposure dose, but the surface exposure dose may be considerably greater than the dose in the absence of the phantom. The total build-up is illustrated by Fig. 14-2, which gives the per cent of air-exposure dose from X-rays commonly used in diagnosis and therapy plotted as a function of depth in the phantom. Figures 14-1 and 14-2 emphasize the need for a knowledge of the energy of the radiation in order to define more completely the radiation hazard. The constant 3.6×10^9 is based on the assumption that each curie corresponds to 3.7×10^{10} disintegrations/sec and that the energy required to produce an ion pair w is 34 ev. If the source emits photons of several energies, the exposure dose D_p is the sum of the exposure doses delivered by photons of each energy.

The **absorbed dose from X- or gamma radiation** is given by the equation

$$D_p = \frac{4.1(\mu - \sigma_a)_m E_1 e^{-\mu_1 X} b_1 B_1 t 10^6}{\rho_m X^2} \quad \text{rads} \quad (14-3)$$

in which D_p is the absorbed dose delivered by a 1-curie point source at a distance X (cm). The term $(\mu - \sigma_a)_m$ is the total coefficient of absorption (cm^{-1}) minus the Compton-scattering coefficient of absorption (cm^{-1}) in medium m , and ρ_m is the density of the medium (g/cc). The medium m in which the absorbed dose is measured in problems of radiation protection is soft tissue, i.e., $\rho_m = 1$. The other terms have the same definitions as in Eq. (14-1). It is to be noted that, when the medium in which the absorbed dose is measured is air at 0°C and 76 cm Hg, $\rho_m = 0.001293$, $(\mu - \sigma_a)_m$ becomes $(\mu - \sigma_a)_a$, and the constant in Eq. (14-3) becomes 3.2×10^9 . If this constant is multiplied by the mass stopping power P_t of tissue relative to air for secondary electrons of energy from about 50 kev to 2 Mev, i.e., $P_t = 1.14$, Eq. (14-3) becomes identical to Eq. (14-1). Therefore, in the usual energy range of X- and gamma radiation, the exposure-dose measurement in roentgens in free air corresponds to the first-collision absorbed-dose measurement in rads in a small element of soft tissue.

The **absorbed dose from a 1-curie point source** of alpha or beta radiation (D_p), delivered in time t (days) at a distance X , well within the range of the radiation, is given by the equation

$$D_p = \frac{4.1tbP}{X^2} \quad \text{rads} \quad (14-4)$$

in which b is the fraction of the disintegrations that result in the emission of an alpha

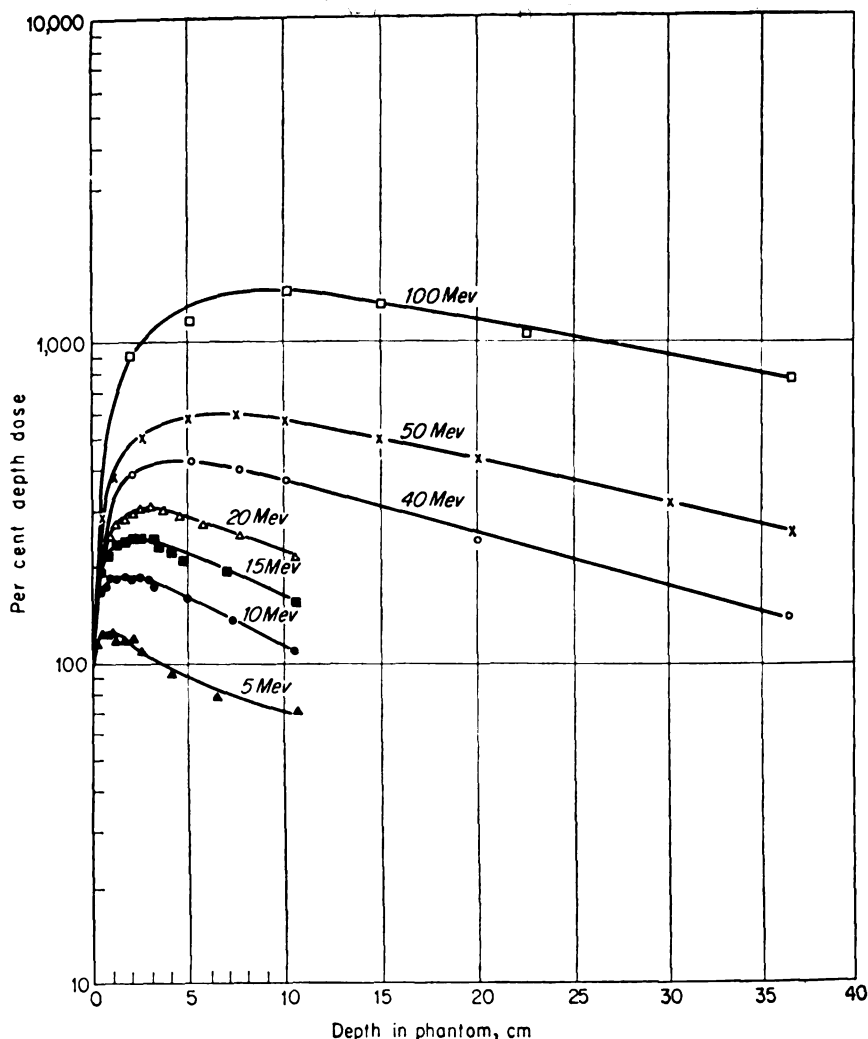


FIG. 14-1. Per cent depth dose vs. depth in phantom. Curves of 100, 50, and 40 Mev by C. B. Braestrup and G. Failla (*unpublished data, 1948*). Curves at 20, 15, 10, and 5 Mev by H. W. Koch, D. W. Kerst, and P. Morrison (*Radiology, vol. 40, pp. 120-127, 1943*).

or beta particle, and P is the mass stopping power in $\text{ev}/(\text{g}/\text{cm}^2)$. The **mass stopping power for alpha radiation** is given by the equation

$$P = \frac{S_a w P_m}{\rho_a} \quad (14-5)$$

in which S_a is the specific ionization in ion pairs per centimeter path in air, w is the electron volts required to produce an ion pair in air, P_m is the mass stopping power

of medium m relative to air, and ρ_a is the density of air at which S_a was measured.

In the case of a source that emits beta radiation, there is considerable scattering, and since the energy of the beta particles ranges from 0 to the maximum beta energy, there is considerable attenuation even at short distances from the source. As a result, this attenuation can be approximated rather well by an exponential for distances well within the maximum range. The stopping power for beta radiation (P_β) is given by the equation

$$P_\beta = \frac{10^6 \mu_m \bar{E} e^{-\mu_m x}}{\rho_m} \quad (14-6)$$

in which P_β is the stopping power [ev/(g/cm²)] of the beta radiation at distance x from the source, $\bar{E}$ is the average initial energy of the beta radiation in Mev, μ_m is the absorption coefficient (cm⁻¹) of the beta radiation in medium m , and ρ_m is the density of medium m .

Sometimes it is convenient to express the exposure dose or absorbed dose in terms of the flux. The exposure dose in terms of flux for X- or gamma radiation is given by the equation

$$D_\gamma = 1.2NEt(\mu - \sigma_s)_a \quad (14-7)$$

in which D_γ is the exposure dose in roentgens measured in free air resulting from X- or gamma radiation of energy E and flux N photons/cm² sec. The term $(\mu - \sigma_s)_a$ is the total minus the Compton-scattering coefficient of absorption in air of the X- or gamma radiation of energy E . The time t is in days. The above equation can be written in terms of absorbed dose in which case

$$D_\gamma = \frac{1.4 \times 10^{-3} NEt(\mu - \sigma_s)_m}{\rho_m} \quad \text{rads} \quad (14-8)$$

Here D_γ is the absorbed dose measured in medium m which has a density ρ_m .

The absorbed dose in terms of flux for ionizing particles, e.g., alpha, beta, protons or heavy ions, is given by the equation

$$D = 1.4 \times 10^{-9} NPt \quad (14-9)$$

in which D is the absorbed dose in rads measured in medium m resulting from a flux of N particles/cm² sec. The stopping power P is given in this case by Eq. (14-5) for any type of ionizing particulate radiation. The time t is in days.

The distribution of X and gamma dose in a tissuelike phantom was indicated by Figs. 14-1 and 14-2. The range of alpha radiation from natural sources of radiation is so short that we may ignore the external dose hazard from pure alpha-emitting sources. In fact, alpha rays with an energy of 7 Mev and beta rays with an energy

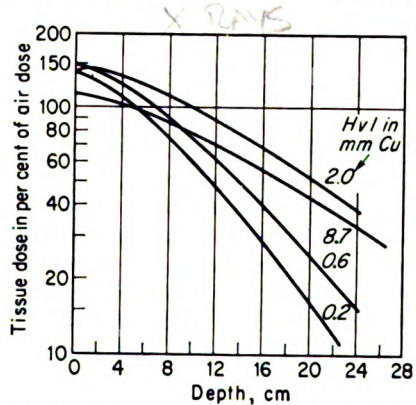


FIG. 14-2. X-ray dose distribution in the main portion of the body.

of 70 kev will just penetrate the minimum thickness (7 mg/cm²) of the protective epidermal layer of skin that surrounds the body. Therefore, from the standpoint of external exposure to alpha and beta sources, one need consider only alpha and beta radiation with energies greater than 7 Mev and 70 kev, respectively. Many sources emit beta radiation with a maximum energy greater than 70 kev.

The **range of beta radiation** is given by the equation

$$R = \frac{1}{\rho} [0.54E_m - 0.13(1 - e^{-3.2E_m})] \quad (14-10)$$

in which R = range (cm) of beta radiation of maximum energy E_m (Mev) in the medium of density ρ (g/cc). Most of the beta particles have energies less than the maximum energy E_m , so only a small fraction reaches the maximum range R . The **average energy $\bar{E}$** (Mev) of beta radiation is given approximately by the equation

$$\bar{E} \simeq 0.33E_m \left(1 - \frac{Z^{1/2}}{50}\right) \left(1 + \frac{E_m^{1/2}}{4}\right) \quad (14-11)$$

in which Z is the atomic number of the beta emitter. This equation can be used to find $\bar{E}$ given in Eq. (14-6). With few exceptions, this equation yields values of $\bar{E}$ with errors of less than 5 per cent.

The **coefficient of absorption of beta particles** in soft tissue [as given in Eq. (14-6)] may be found approximately from the equation

$$\mu_t \approx 16E_m^{-1.6} \quad (14-12)$$

in which μ_t (cm⁻¹) is the coefficient of absorption in tissue of beta particles of maximum energy E_m . This equation is in error less than 20 per cent in the energy range 0.1 to 2 Mev. The **specific ionization** (ion pairs/cm) in air S_a of beta particles [as given in Eq. (14-5)] may be obtained approximately from the equation

$$S_a \approx 33 + 63E_m^{-0.9} \quad (14-13)$$

This equation is in error less than about 10 per cent in the energy range 0.05 to 2 Mev. In the case of alpha particles in air

$$S_a \approx 9.5 \times 10^4 E^{-0.74} \quad (14-14)$$

This equation is in error less than about 5 per cent in the energy range 3 to 20 Mev.

The relative mass stopping power P_m for soft tissue [as given in Eq. (14-5)] is an insensitive function of energy of the ionizing particle. It ranges from 1.16 to 1.13 for beta particles of maximum energy 10 kev and 2 Mev, respectively, and from 1.22 to 1.20 for alpha particles of energy 2 and 4 Mev, respectively. The electron volts required to produce an ion pair w may be set equal to 34 for X-rays, gamma rays, and beta particles in air, and 35 for alpha particles in air.

The neutron is a particle which produces ionization not directly but indirectly. **Fast neutrons** produce recoil ions, and these, in turn, produce ionization by two mechanisms (Snyder, W. S., and J. Neufeld, On the Energy Dissipation of Moving Ions in Tissue, *Oak Ridge Natl. Lab. Rept.* ORNL-1083, Nov. 14, 1951), (1) **electronic collision** or the interaction of the ions with electrons of the tissue atoms and (2) **nuclear collisions** or the interaction of the ions with the nuclei. At high energies

essentially all the energy of the ion is dissipated through electronic collisions, but at low energies the energy losses due to nuclear collisions become significant. Some of the neutrons escape from the body while others continue to undergo collisions until they are thermalized. Thermalized fast neutrons or **thermal neutrons** dissipate their energy in soft tissue primarily by two processes: (1) neutron capture in hydrogen, $\text{H-1}(n,\gamma)\text{H-2}$, and (2) neutron capture in nitrogen, $\text{N-14}(n,p)\text{C-14}$.

In determining the **absorbed dose from neutrons**, one is usually concerned with the total absorbed dose, which includes both the primary energy loss and the contribution from multiple scattering in the phantom. However, in some special cases, such as instrument calibration or the estimation of the absorbed dose delivered to small targets, the first-collision absorbed dose is desired.

The **first-collision absorbed dose for fast neutrons** is given by the equation

$$D = 1.4 \times 10^{-3} N E t \Sigma f_i \sigma_i e_i \quad (14-15)$$

in which D is the absorbed dose in rads, N is the neutron flux in neutrons/cm² sec, E is the energy per neutron in Mev, t is the time in days, and $\Sigma f_i \sigma_i e_i$ is the fractional energy loss per g/cm³ of tissue. When the absorbed dose is delivered to soft tissue,

$$f_i = \frac{6.02 \times 10^{23} F_i}{A_i} \quad (14-16)$$

in which f_i is the atoms of the i th type per gram of tissue, F_i is the fraction by weight of the i th-type element in tissue, and A_i is the atomic weight of the i th element in tissue. Using this equation, one obtains the atoms per gram of soft tissue for hydrogen, oxygen, carbon, and nitrogen as 5.98×10^{22} , 2.75×10^{22} , 6.02×10^{21} , and 1.72×10^{21} , respectively.

The term e_i , the **fraction of energy lost per neutron collision** with atoms of soft tissue, is given by

$$e_i = \frac{2M_i m}{(M_i + m)^2} \quad (14-17)$$

in which m/M_i is the ratio of the mass of the neutron to the mass of the i th atom of soft tissue. In this equation it is assumed there is isotropic scattering—a reasonable approximation for neutrons of energy less than 10 Mev. The values for e_i for hydrogen, oxygen, carbon, and nitrogen are 0.500, 0.112, 0.143, and 0.125, respectively.

The cross section σ_i of the i th atom in cm² must be obtained for the neutrons of each energy considered.

The **first-collision absorbed dose for thermal neutrons** (in rads) is given by the equation

$$D = 1.4 \times 10^{-3} N t f_N \sigma_N E_N = 2.0 \times 10^{-4} N t \quad (14-18)$$

in which E_N is the energy delivered to tissue as a result of neutron capture by the nitrogen atom. The neutron capture in hydrogen need not be considered in this case because in the $\text{H-1}(n,\gamma)\text{H-2}$ reaction essentially all the gamma energy escapes from the small tissue element.

Table 14-1 gives values of first-collision absorbed dose in terms of neutrons/cm²/rad as a function of neutron energy. Corresponding values are given of

Table 14-1. Flux of Neutrons Corresponding to 1 Rad or 1 Rem

Neutron energy, Mev	First-collision calculation of flux					Multiple-collision calculation of flux †			
	$n_i/\text{cm}^2/\text{rad}$	$n/\text{cm}^2/\text{rad}^*$	Using fixed RBE		Using NBS Handbook 59 values of RBE	$n/\text{cm}^2/\text{rad}$	Using fixed RBE	Using NBS Handbook 59 values of RBE	
			$n/\text{cm}^2/\text{rem}$	$n/\text{cm}^2/\text{rem}^*$	$n/\text{cm}^2/\text{rem}$				$n/\text{cm}^2/\text{rem}$
10	1.78×10^8	1.78×10^8	1.55×10^7	1.55×10^7	2.86×10^7	2.86×10^7	1.4×10^8	1.3×10^7	2.2×10^7
5	2.23×10^8	2.23×10^8	2.04×10^7	2.04×10^7	3.45×10^7	3.45×10^7	1.8×10^8	1.7×10^7	2.5×10^7
4	2.33×10^8	2.33×10^8	2.10×10^7	2.10×10^7	3.33×10^7	3.33×10^7	2.1×10^8	1.9×10^7	2.7×10^7
3	2.72×10^8	2.72×10^8	2.54×10^7	2.54×10^7	3.77×10^7	3.77×10^7	2.3×10^8	2.2×10^7	2.9×10^7
2	3.23×10^8	3.23×10^8	3.01×10^7	3.01×10^7	4.00×10^7	4.00×10^7	2.5×10^8	2.4×10^7	2.9×10^7
1	4.05×10^8	4.05×10^8	4.05×10^7	4.05×10^7	3.85×10^7	3.85×10^7	2.6×10^8	2.5×10^7	2.9×10^7
0.5	6.13×10^8	6.13×10^8	5.56×10^7	5.56×10^7	5.56×10^7	5.56×10^7	4.4×10^8	4.2×10^7	4.5×10^7
0.1	1.61×10^9	1.61×10^9	1.54×10^8	1.54×10^8	1.25×10^8	1.25×10^8	1.0×10^9	1.4×10^8	1.3×10^8
0.01	1.06×10^{10}	1.04×10^{10}	1.05×10^9	1.05×10^9	9.09×10^8	9.09×10^8	1.8×10^9	6.9×10^8	6.5×10^8
10^{-4}	7.63×10^{11}	8.13×10^{10}	7.52×10^{10}	4.12×10^{10}	6.67×10^{10}	3.85×10^{10}	1.4×10^9	7.3×10^8	7.2×10^8
2.5×10^{-3}	4.54×10^{10}	1.38×10^9	4.50×10^9	1.09×10^9	4.35×10^9	1.08×10^9	3.0×10^9	1.0×10^9	1.0×10^9

* The values of flux for first-collision calculation, as given in columns 3, 5, and 7, were obtained by assuming all the gamma energy produced in the element of tissue is absorbed in the element and contributes to the absorbed dose.

† These values of flux were obtained by W. S. Snyder by applying a Monte Carlo calculation to obtain the absorbed dose at various depths in a large-tissue phantom 30 cm thick. The values of flux correspond to the absorbed dose at a position in the phantom where it is a maximum.

neutrons/cm²/rem for two cases: (1) **fixed values of relative biological effectiveness, RBE**, and (2) **variable values of RBE** following the same functional relationship between RBE and linear energy transfer, LET, as is given in the NBS Handbook 59 ("Permissible Dose from External Sources of Ionizing Radiation," GPO, Sept. 24, 1954). The fixed values assumed for RBE in calculating the flux corresponding to 1 rem in Table 14-1 are (RBE)_e = 1 for gamma rays or secondary electrons, (RBE)_P = 10 for protons, and (RBE)_H = 20 for heavy-recoil ions. Also, neutrons/cm²/rad and neutrons/cm²/rem are given for the absorbed dose (for fixed RBE and for variable RBE) at the position in the phantom where the absorbed dose is a maximum. These values for flux based on multiple collision in the phantom were obtained by W. S. Snyder (NBS Handbook 63, "Protection against Neutron Radiation up to 30 Million Volts") using a Monte Carlo type of calculation. The reader is referred to this publication for a detailed breakdown of the absorbed dose contributed by the secondary electrons, protons, and heavy ions at various depths in the tissue phantom and for the various neutron energies.

A type of exposure from an external source which is of special interest to those concerned with radiation hygiene is in the case of an **individual immersed in a large body of radioactive gas or water**. The absorbed dose D in rads is given by the equation

$$D = 51 \frac{C \bar{E} P_t}{\rho_m} \quad (14-19)$$

in which C is the $\mu\text{c/cc}$ in the surrounding medium, $\bar{E}$ (Mev) is the average energy emitted per disintegration (alpha, beta, or gamma), P_t is the stopping power of the medium (tissue) in which the absorbed dose is measured relative to the stopping power in the surrounding medium (air or water), ρ_m is the density (g/cc) of the surrounding medium, and t is the time of exposure in days. In this equation it is assumed the medium containing the radioactive material is large compared with the range of the alpha or beta particles. In the case of a gamma emitter it is assumed that $(1 - e^{-\mu X}) \cong 1$, in which X is the radius of the surrounding medium and μ is the total coefficient of absorption of the radiation in the medium. For example, this equation can be used to obtain the maximum absorbed dose from a large cloud of radioactive gas such as **Argon-41** or **Xenon-133**. In this case the absorbed dose from the large cloud of noble gas surrounding the body is far in excess of the absorbed dose from the noble gas contained in the lungs. In general, this equation overestimates the absorbed dose received by a person by at least a factor of 2 with the possible exception of the absorbed dose to the ear which may be irradiated from slightly more than one-half the solid angle. The equation probably gives a good estimate of the absorbed dose received by fish eggs suspended in a body of water that is uniformly contaminated with radioactive material. Equation (14-19) can be applied to calculate the **maximum permissible concentration, MPC**, of a noble gas in air by rewriting it in the form

$$\text{MPC} = \frac{2.8 \times 10^{-3} \dot{D} \rho_a}{\bar{E} P_t} \approx \frac{10^{-6}}{\bar{E}} \quad (14-20)$$

in which MPC is the $\mu\text{c/cc}$ of air that will deliver an absorbed dose rate of $\dot{D}$ rads/week to a small element of tissue. The last part of the equation is obtained

by placing $\dot{D} = 0.3$ rad/week, $\rho_a = 0.001293$ and $P_t = 1.14$. In most cases, the MPC values can be doubled because the radiation to the body is seldom from more than one-half the solid angle.

ABSORBED DOSE FROM INTERNAL SOURCES

According to the strict definition of the roentgen unit, one should not apply the roentgen to measure the dose inside the body. Therefore, only the rad and rem units of absorbed dose will be used to express **absorbed dose from internal sources**. Some years ago, and especially in the early period of the atomic-energy program, 1942-1949, considerable difficulty and confusion existed when one attempted to compare calculations on absorbed dose from radionuclides contained inside the body because each person making the calculations used his own preferred values for the so-called **standard man**. As a result, it was decided at the Chalk River, Canada, Conference in 1949 to define the standard man. Although some of the values chosen for this standard man may not be very representative for an individual, e.g., one who is very fat or one who is very small and thin, they nevertheless have served the very useful function of reducing all these calculations to a common denominator such that the various calculations are directly comparable. Tables 14-2, 14-3, and 14-4 list characteristics of the standard man as listed in the handbook of the International Commission on Radiological Protection, ICRP Handbook (Supplement 6 "Recommendations of the International Commission on Radiological Protection,"

Table 14-2. Average Chemical Composition of the Adult Human Body

Element	% by weight	Approx amount in a 70-kg man, g
Oxygen, O.....	65.0	45,500
Carbon, C.....	18.0	12,600
Hydrogen, H.....	10.0	7,000
Nitrogen, N.....	3.0	2,100
Calcium, Ca.....	1.5	1,050
Phosphorus, P.....	1.0	700
Sulfur, S.....	0.25	175
Potassium, K.....	0.2	140
Sodium, Na.....	0.15	105
Chlorine, Cl.....	0.15	105
Magnesium, Mg....	0.05	35
Iron, Fe.....	0.006	4
Manganese, Mn.....	0.00003	0.02
Copper, Cu.....	0.0002	0.1
Iodine, I.....	0.00004	0.03

Table 14-3. Mass and Effective Radius of Organs of the Adult Human Body *

(Standard man)

Part of body	Mass <i>m</i> , g	% of total body †	Effective radius <i>X</i> , cm
Total body †.....	70,000	100	30
Muscle.....	30,000	43	30
Skin and subcutaneous tissue †.....	6,100	8.7	0.1
Fat.....	10,000	14	20
Skeleton:			
Without bone marrow.....	7,000	10	5
Red marrow.....	1,500	2.1	
Yellow marrow.....	1,500	2.1	
Blood.....	5,400	7.7	
Gastrointestinal tract †.....	2,000	2.9	30
Contents of gastrointestinal tract:			
Lower large intestine.....	150		5
Stomach.....	250		
Small intestine.....	1,100		
Upper large intestine.....	135		
Liver.....	1,700	2.4	10
Brain.....	1,500	2.1	
Lungs (2).....	1,000	1.4	10
Lymphoid tissue.....	700	1.0	
Kidneys (2).....	300	0.43	7
Heart.....	300	0.43	7
Spleen.....	150	0.21	7
Urinary bladder.....	150	0.21	
Pancreas.....	70	0.10	5
Salivary glands (6).....	50	0.071	
Testes (2).....	40	0.057	3
Spinal cord.....	30	0.043	1
Eyes (2).....	30	0.043	0.25
Thyroid gland.....	20	0.029	3
Teeth.....	20	0.029	
Prostate gland.....	20	0.029	3
Adrenal glands or suprarenal (2)....	20	0.029	3
Thymus.....	10	0.014	
Miscellaneous (blood vessels, cartilage, nerves, etc.).....	390	0.56	

* Lisco, Hermann, *The Standard Man, Argonne Natl. Lab. Rept. ANL-4253*, November, 1948–February, 1949, p. 96, and Cook, M. J., *A Survey Report of the Characteristics of the Standard Man*, September, 1948, were used as the principal sources of reference in the original selection of values given in this table.

† Does not include contents of gastrointestinal tract.

‡ The mass of the skin alone is taken as 2,000 g.

Table 14-4. Characteristics of the Standard Man

	<i>cc/day</i>
Water intake in food	1,000
Water intake as fluids	1,200
Water intake in food and fluids	2,200
Water of oxidation	300
Total water consumption per day	2,500
Water excretion as sweat	600
Water excretion from lungs	300
Water excretion by feces	200
Water excretion by urine	1,400
Total water excretion per day	2,500
Air inhaled during 8 hr workday	10^7
Air inhaled during 16 hr not at work	10^7
Total air inhaled per day	2×10^7
CO ₂ in alveolar air, 5.5%	
Interchange area of lungs	50 sq m
Area of upper respiratory tract, trachea, bronchials	20 sq m
Total surface area of respiratory tract	70 sq m
Total water in body, 43 kg	
Average life span of man, 70 years	
Occupational exposure of man: 8 hr/day, 40 hr/week, and 50 weeks/year	
Minimum thickness of epidermis, 0.07 mm, and depth of blood-forming organs, 5 cm	

Particulates in Respiratory Tract

Retention of particulate matter in the lungs depends on many factors, such as the size, shape, and density of the particles, the chemical form and whether or not the person is a mouth breather; however, when specific data are lacking it is assumed the distribution is as follows:

Distribution	Readily soluble compounds, %	Other compounds, %
Exhaled	25	25
Deposited in upper respiratory passages and subsequently swallowed	50	50
Deposited in the lungs (lower respiratory passages)	25 (this is taken up into the body)	25 *

* Of this, half is eliminated from the lungs and swallowed in the first 24 hr, making a total of $62\frac{1}{2}\%$ swallowed. The remaining $12\frac{1}{2}\%$ is retained in the lungs with a half-life of 120 days, it being assumed that this portion is taken up into body fluids.

ICRP Handbook, The British Institute of Radiology, 1955, ICRP and NCRP Handbook 1959 revisions).

Table 14-2 gives the per cent by weight of some of the more common elements in the body and the weight of these elements in a 70-kg man. Similar information is needed for the trace elements and for the distribution of the elements in the various body organs. Although most of these data are lacking, some of them are being made available by spectrographic analysis of human tissue by Tipton *et al.* (1959 edition of the "Internal Dose Report of the ICRP"). Table 14-3 gives the mass of the various body organs, the per cent of each organ of the total body weight, and the effective radius of each organ. The latter figure is used in estimating the fraction of the gamma energy per disintegration of the radionuclide that is received by the organs. Table 14-4 lists factors of water intake and excretion, breathing rates, dust retentions, etc.

One of the first problems in determining the hazard from radionuclides taken into the body is to select the **critical organ** of the body. This organ is defined as that body organ receiving the radionuclide that results in the greatest damage to the body. This is almost invariably the organ receiving the greatest damage or the one with the greatest concentration, although this does not necessarily follow. The **maximum permissible body burden** q of any radionuclide that is primarily localized in the bone is that amount contained in the total body which will result in an absorbed dose rate to the bone that is equivalent to the absorbed dose rate delivered to the bone as a consequence of 0.1 μg of Ra-226 contained in the total body. In this case q is given by the equation

$$q = \frac{0.1(0.99)111}{f_2 \Sigma E(\text{RBE})N} = \frac{11}{f_2 \Sigma E(\text{RBE})N} \quad (14-21)$$

in which q is the μc of the bone-seeking radionuclide in the total body that will deliver to the bone the same absorbed dose (rems) as 0.1 μg of Ra-226. It is assumed that 99 per cent of the Ra-226 and radium daughter products that are in the body are in the bone. f_2 is the fraction in the critical body organ (bone in this case) of that in the total body, and $\Sigma E(\text{RBE})N$ is the effective energy per disintegration of the radionuclide; in the case of Ra-226 and its daughter products $\Sigma E(\text{RBE})N = 110$ Mev. The energy per disintegration E is multiplied by the relative biological effectiveness RBE and by the non-uniform distribution factor N . Values in the 1958 ICRP Handbook are $(\text{RBE})_H = 20$ for heavy ions, $(\text{RBE})_\alpha = 10$ for alpha, and $(\text{RBE})_e = 1$ for secondary electrons. In NBS Handbook 52 ("Maximum Permissible Amounts of Radioisotopes in the Human Body and Maximum Permissible Concentrations in Air and Water," GPO, Mar. 20, 1953), which was printed a year earlier than the ICRP Handbook, $(\text{RBE})_H$ was not used, $(\text{RBE})_\alpha = 20$, and $(\text{RBE})_e = 1$. Some experiments (Langham, W. H., and John Storer, "Quantitative Biological Methods for Studying Radiation Effects in Mammals," Paper presented at the Radiation Research Society Meeting, New York, May 18, 1955) have indicated that the **RBE for alpha particles and fast neutrons** may be closer to 2 than to 10 for acute biological damage. No conclusive evidence is available at this time (1959) to indicate that the RBE for these two types of particulate radiation should be reduced for chronic exposure. In the ICRP Handbook the factor N is set equal to 5 for alpha and beta radiation from all bone-

seeking radionuclides with the exception of Ra-226. This was done rather arbitrarily because certain of the radionuclides, such as Sr-89, Sr-90, and Pu-239, gave experimental evidence of producing about five times the damage per rad as Ra-226. Actually N should be a "relative hazard" factor since it probably includes several factors such as (1) nonuniform distribution, (2) radiosensitivity, and (3) essentialness of the body organ.

The **maximum permissible body burden q for critical organs other than the bone** is determined from the equation

$$q = \frac{2.8 \times 10^{-3} m \dot{D}}{f_2 \Sigma E(\text{RBE})N} = \frac{8.4 \times 10^{-4} m}{f_2 \Sigma E(\text{RBE})N} \quad (14-22)$$

in which q is the μC in the critical organ of mass $m(\text{g})$ that will deliver an absorbed dose rate, $\dot{D} = 0.3 \text{ rem/week}$; * the other terms are defined above.

The preceding equations can be used to determine the maximum permissible body burden q or the maximum permissible burden qf_2 in the critical body organ for a radionuclide. However, when one wishes to relate q or qf_2 to the amount of the radionuclide taken into the body, the problem becomes more involved since we must know the fraction of that taken into the body that arrives in the critical body organ and the rate of elimination from the critical body organ. It has been recognized that no simple pattern of uptake, distribution, and elimination from the body can be expected to represent accurately the absorbed dose delivered by the burden qf_2 of the radionuclide in the critical body organ. In fact, the biological data available to be used in the calculation of this absorbed dose are so limited for the various radionuclides that many simplifying assumptions are both necessary and justified. In most cases the amount in the critical body organ following a single intake of the radionuclide can be expressed adequately by the equation

$$Q_t = Q_1 e^{-0.693t/T_1} + Q_2 e^{-0.693t/T_2} + \dots \quad (14-23)$$

in which Q_t is the burden of the radionuclide in the critical body organ at any time t after arrival in the critical organ. Q_1, Q_2 , etc., are components with various amounts of fixation or with various half-lives T_1, T_2 , etc., such that at time $t = 0$, $Q_0 = Q_1 + Q_2 + \dots$. Of course, the more terms taken the closer the theory approximates the experimental data; however, in most cases the experimental data are so scanty that only one exponential term need be used. In some cases Eq. (14-23) can be approximated best by means of a power function (Langham, Wright, H., *Determination of Internally Deposited Radioactive Isotopes from Excretion Analyses, Ind. Hyg. Quart.*, vol. 17, no. 3, pp. 305-314, September, 1956. Healy, J. W., *The Status of Bioassay Techniques for the Evaluation of Internal Exposure from Radiation Isotopes, Proc. Health Phys. Soc.*, Ann Arbor, Mich., June 25, 26, and 27, 1956). In this case

$$Q_t = Q_0 t^{-K} \quad (14-24)$$

and K is a different constant for each radionuclide. This equation cannot be applied to small values of t , and in many cases insufficient data are available to obtain suit-

* $\dot{D} = 0.1 \text{ rem/week}$ for gonads and total body and 0.6 rem/week for skin and thyroid (1959 ICRP and NCRP editions).

able values of the terms K or Q_0 . Although it is probable that in the near future use will be made of the power-function equation (14-24) in obtaining values of q and of MPC for some of the radionuclides, at present all values given in the handbooks on internal dose (NBS Handbook 52 and the ICRP Handbook) make use of a single exponential as given by the first term in Eq. (14-23). The **maximum permissible concentration (MPC)** of a radionuclide in air for **continuous exposure** is given by the equation

$$(\text{MPC})_a = \frac{3.5 \times 10^{-8} q f_2}{T f_a (1 - e^{-0.693t/T})} \quad (14-25)$$

in which $(\text{MPC})_a$ is the $\mu\text{c/cc}$ of air that will result in the deposition of the maximum permissible burden $q f_2$ in the critical body organ in a period of exposure of t (days) for a radionuclide with an effective half-life of T (days). The time t is set equal to 50 years for occupational exposure at the maximum permissible concentration $(\text{MPC})_a$. In this period of exposure equilibrium is reached for most of the radionuclides with the exception of a few such as Ra-226 and Pu-239. The constant f_a is the fraction of the inhaled material that arrives initially in the critical body organ. The **effective half-life** T is given by the equation

$$T = \frac{T_b T_r}{T_b + T_r} \quad (14-26)$$

in which T_b is the biological half-life and T_r is the radioactive half-life.

In the case of **ingestion**,

$$(\text{MPC})_w = \frac{3.1 \times 10^{-4} q f_2}{T f_w (1 - e^{-0.693t/T})} \quad (14-27)$$

in which f_w is the fraction of the ingested material that arrives in the critical body organ.

A case of special interest is the calculation of the values of MPC when the **gastro-intestinal tract is the critical body organ**. When the material is ingested,

$$(\text{MPC})_w^{\text{GI}} = \frac{7.6 \times 10^{-7} m' G}{H f_w \Sigma E(\text{RBE})} \quad (14-28)$$

in which m' is the mass of the material in a particular section of the gastrointestinal tract, H is the time (days) in this section, f_w is the fraction of that ingested that arrives in this section, and $\Sigma E(\text{RBE})$ is the effective energy given off for each disintegration of the radionuclide. The term G is given by the equation

$$G = \frac{0.693H}{(e^{-0.693h_0/T_r} - e^{-0.693h_1/T_r})T_r} \quad (14-29)$$

in which h_0 is the time (days) of arrival in the section of the gastrointestinal tract following ingestion, and h_1 is the time (days) of departure, i.e., $H = h_1 - h_0$. When $T_r \gg h_0$ and h_1 , $G = 1$.

In the case of **inhalation of radioactive dust**, a large fraction of the material is swallowed. As seen from Table 14-4, it is assumed in the case of relatively **soluble material** 50 per cent of the inhaled material is held up in the upper respiratory tract; then it is swallowed and 25 per cent reaches the lower respiratory tract. In the case

of relatively **insoluble material**, half of this, or 12.5 per cent, is brought up by the cilia of the bronchi and swallowed within 24 hr so that the total fraction swallowed of that inhaled is assumed to be 0.62. Thus,

$$(\text{MPC})_a^{\text{GI}} = \frac{8.4 \times 10^{-11} m' G}{H f_a \Sigma E(\text{RBE})} \quad (14-30)$$

The fraction going from the gastrointestinal tract to the blood is given as f_1 . It follows to a close approximation that $f_w = 1 - f_1$ and $f_a = 0.62(1 - f_1)$ as applied in Eqs. (14-28) and (14-30), respectively. Since the material in the gastrointestinal tract resides in the lower large intestine much longer than in other sections and the mass of material is reduced to a minimum by the time it reaches this section, the values of $(\text{MPC})_w^{\text{GI}}$ and $(\text{MPC})_a^{\text{GI}}$ are usually based on the lower large intestine as the critical body organ. In this case $m' = 150$ g, $h_0 = 1\frac{3}{4}$ days, $h_1 = 3\frac{1}{4}$ days, and $H = 1\frac{8}{24}$ days.

The values of $(\text{MPC})_a$ and $(\text{MPC})_w$ in NBS Handbook 52 and the ICRP Handbook apply only to continuous exposure. These values were obtained by means of Eqs. (14-25) and (14-27). The values of $(\text{MPC})_a^{\text{GI}}$ and $(\text{MPC})_w^{\text{GI}}$, as given in Eqs. (14-28) and (14-30), are based on an exposure rate of 0.3 rem/week to the intestinal wall. Equations (14-28) and (14-30) can be written in terms of the absorbed dose D delivered to the critical portion of the **gastrointestinal tract** as follows:

$$D = \frac{26 f I_0 \Sigma E(\text{RBE}) H}{m' G} \quad (14-31)$$

in which D is the most probable absorbed dose in rems, i.e., from one-half the solid angle, delivered to the critical section of the gastrointestinal tract as a result of a single intake of I_0 (μc) of the radionuclide. In most cases the lower large intestine is the critical section of the gastrointestinal tract, in which case $m' = 150$ g, $h_0 = 1\frac{3}{4}$ days, $h_1 = 3\frac{1}{4}$ days, and $H = 1\frac{8}{24}$ days. The **absorbed dose to the lower large intestine** is given by the equation

$$D = \frac{0.13 f I_0 \Sigma E(\text{RBE})}{G} \quad (14-32)$$

in which D is the most probable absorbed dose in rems, i.e., from one-half the solid angle, received by the wall of the lower large intestine as the result of the intake of I_0 (μc) of a radionuclide. The term $f = (1 - f_1)$ in the case of ingestion and $f = 0.62(1 - f_1)$ in the case of inhalation. When $T_r \gg 1$ day, e.g., $T_r \geq 1$ week, G may be set equal to 1.

The above equations that refer to the gastrointestinal tract as the critical body tissue, i.e., Eqs. (14-28) to (14-32), can be applied to either chronic or **acute exposure** because the maximum time the material remains in any section of the gastrointestinal tract is only 18 hr. When other organs of the body, e.g., liver, spleen, kidney, or bone, are the critical organs, Eqs. (14-25) and (14-27) can be applied to them only for chronic exposure. In this case official MPC values are given in the handbooks, i.e., NBS Handbook 52 and the ICRP Handbook, only for continuous exposure. For the usual occupational exposures of 40 hr/week these MPC values in the handbooks can be increased by a factor of 3. In case of **single or acute ex-**

posure, the absorbed dose may be obtained from the equation

$$D = \frac{74f_{wa}I_0T\Sigma E(\text{RBE})N}{m} \frac{1 - e^{-0.693t/T}}{1} \quad (14-33)$$

in which D is the absorbed dose (rems) to any organ of the body (other than the gastrointestinal tract) as a result of I_0 (μc) taken into the body in a single event. It is assumed that the fraction f_{wa} reaches the critical body organ of mass m (g) and that it has an effective half-life T (days). For ingestion $f_{wa} = f_w$ and for inhalation $f_{wa} = f_a$; values of these constants are given for the various radionuclides in Handbook 52 and the ICRP Handbook. The term $\Sigma E(\text{RBE})N$ is the effective energy per disintegration delivered to the critical organ, and t is the period of exposure in days. The maximum permissible absorbed dose D has not been officially defined by the NCRP or the ICRP. However, it is rather common practice to set it equal either to 0.3 rem for a time $t = 1$ week or to 3 rems for a period $t = 13$ weeks.

In case a person is **exposed to several sources of ionizing radiation simultaneously**, e.g., external exposure to X-rays while at the same time internal exposure to one or more radionuclides, the procedure in selecting the values of MPC is to set them at such a level that the total absorbed dose rate $\dot{D}$ or the total absorbed dose D to any body organ will not exceed 0.3 rem in a week.* In making this summation, one should keep in mind that the MPC values given by Eqs. (14-20), (14-25), (14-27), (14-28), and (14-30) correspond to a dose rate $\dot{D} = 0.3$ rem/week. Illustrations of how to make this compilation are given in the ICRP Handbook.

One of the best ways to estimate the body burden of a radionuclide is by an **examination of the body excretions**. If a person has been exposed to a radionuclide over a considerable period of time such that the maximum permissible body burden q is attained, the total amount eliminated (by urine, feces, etc.) in the time interval $t_2 - t_1$ is

$$U = \frac{qf_2T}{T_bC} (e^{-0.693t_1/T} - e^{-0.693t_2/T}) \quad (14-34)$$

It is assumed that the person is removed from further exposure at time $t = 0$ and that all excretions are collected beginning at time $t = t_1$ and terminating at time $t = t_2$. U is the μc excreted in the interval $t_2 - t_1$, T_b and T are the biological and effective half-lives, respectively, and C is the ratio of the amount eliminated from the critical body organ to the total amount eliminated. Frequently the term C is unknown, but this difficulty can be circumvented partly by placing $f_2/C = 1$ and letting T and T_b apply to the effective and biological half-lives, respectively, of the entire body rather than to the critical organ. The equation to find the total fecal elimination of a radionuclide during the few days (not less than 31 hr) following a maximum permissible single intake I_0 is

$$U = fI_0e^{-0.693t/T}, \quad (14-35)$$

The proper value of I_0 may be obtained from Eqs. (14-31), (14-32), or (14-33), whichever gives the smallest value. f is equal to $(1 - f_1)$ for ingestion and

* $\dot{D} = 0.1$ rem/week for gonads and total body and $\dot{D} = 0.6$ rem/week for skin and thyroid (1959 ICRP and NCRP editions).

$0.62(1 - f_1)$ for inhalation. T_r is the radioactive half-life of the radionuclide in days and U , the total μc eliminated by the feces, is measured at time t following the single intake of the radionuclide.

COMMENTS ON THE CRITICAL-ORGAN CONCEPT

The validity of the concept of **critical organ** in the sense used in calculating maximum permissible body burden has been considered in some detail by Stannard (*Univ. Rochester Atomic Energy Project Rept. UR-402*, June 30, 1955, *Proc. Intern. Conf. Peaceful Uses Atomic Energy, Geneva*, vol. 13, p. 205, 1956). Obviously choosing the organ with the highest concentration of radioactive material, as is usually done, disregards other possible criteria such as relative radiosensitivity, evidence of histopathology, or the relative importance of damage to that particular organ to the body as a whole. Also, there is evidence that toxicity, for some elements at least, may be related more to total body content than to the amounts in any given tissue (Stannard, Della Rose, and Thomas, *Radiation Research*, vol. 5, abstract 95, November, 1956). Thus the concept as utilized may be theoretically unsound in some instances, though not necessarily in all.

On the other hand, as shown by Stannard, the maximum permissible body burdens calculated by the use of several other criteria seldom vary by more than a factor of 10 in most instances. Also, the figure obtained by allowing no organ to receive more than 0.3 rem/week, regardless of relative radiosensitivity, is on the conservative side. Therefore, if it is relatively easy to meet the maximum air and water levels associated with the lower maximum body burden, an appreciable safety factor is probably involved. On the other hand, when crucial matters of cost, equipment design, or logistics are involved, closer examination of the validity of any given criterion would be in order. Thus, the importance of validating the critical-organ concept in detail for each given situation will increase as the economic problems of potential nuclear power assume greater importance.

General References

"Permissible Dose from External Sources of Ionizing Radiation," NBS Handbook H-59, GPO, Sept. 24, 1954.

Snyder, W. S., The Variation of Neutron Dose with Neutron Energy and Geometry, *Proceedings of the Ann Arbor Meeting of the Health Physics Society*, Ann Arbor, Mich., June 25-27, 1956.

"Protection against Neutron Radiation up to 30 Million Electron Volts," NBS Handbook 63, GPO, November 22, 1957.

Supplement 6, "Recommendations of the International Commission on Radiological Protection," ICRP Handbook, The British Institute of Radiology, 1955.

Tipton, I. H., M. J. Cook, E. G. Struxness, and K. Z. Morgan, Tentative title, "Spectrographic Analysis of Human Tissue for Trace Elements" (in process of being written).

"Maximum Permissible Amounts of Radioisotopes in the Human Body and Maximum Permissible Concentrations in Air and Water," NBS Handbook 52, GPO, Mar. 20, 1953.

Langham, W. H., and John Storer, "Quantitative Biological Methods for Studying Radiations Effects in Mammals," Paper presented at the Radiation Research Society Meeting, New York, May 18, 1955.

Langham, Wright H., Determination of Internally Deposited Radioactive Isotopes from Excretion Analyses, *Ind. Hyg. Quart.*, vol. 17, no. 3, pp. 305-314, September, 1956.

Healy, J. W., The Status of Bioassay Techniques for Evaluation and of Internal Exposures from Radioactive Isotopes, *Proc. Health Phys. Soc. Meeting*, Ann Arbor, Mich., June 25-27, 1956.

"Internal Dose Report of the ICRP" (1959 edition).

Section 15

NUCLEAR SAFETY

By

DIXON CALLIHAN, *Oak Ridge National Laboratory.*

JOHN OZEROFF, *General Electric Company Vallecitos Atomic Laboratory,
Pleasanton, California*

HUGH C. PAXTON, *Los Alamos Scientific Laboratory.*

C. L. SCHUSKE, *Rocky Flats Plant, Dow Chemical Company.*

Factors Determining Criticality.....	15-3
Instrumentation.....	15-3
Design Criteria.....	15-4
Recommended Controls.....	15-4

NUCLEAR SAFETY

Dixon Callihan, John Ozeroff, Hugh C. Paxton, and C. L. Schuske

Much of the material upon which the contents of this chapter are based has been under security classification until very recently, and even now only part of the original data has been published in the open literature. The General References on page 15-10 include classified and unclassified AEC reports, of interest to those readers to whom they may be available, which either give the data or refer, in turn, to the original sources.

The expansion of the nuclear-energy program into industry with the attendant needs for fuel preparation and reprocessing brings with it the recognition of a unique problem in safety which is the subject of this chapter. The radiation energy which may be released in an uncontrolled nuclear chain reaction in chemical operations can be lethal. The thermal energy released simultaneously can cause damage to equipment and can disperse radioactive contaminants in the immediate area of the reaction. For these and other reasons it is imperative that all operations with fissionable materials be free from accidental and unscheduled critical, or chain-reacting, conditions; so the design of equipment and processes must include controls necessary to prevent potentially hazardous accumulations. The term **nuclear safety** has been applied to the study of the specifications necessary to prevent such critical conditions from developing in process operations with the fissionable isotopes and in concomitant activities, such as transportation and storage.

The problem divides itself into three parts. One is scientific and includes the measurement of basic nuclear properties, their application in the calculation of critical parameters, and a comparison with the results of experiments performed with critical assemblies operated at essentially zero power. If accurate data and acceptable methods of reactor analysis were available, the specifications necessary for nuclear safety could be calculated. But since such is not possible, except in cases of simple geometry and of particular materials, the recommendations included in this chapter are based almost entirely on experimental data. Another part of the problem is of an engineering type. The expected location and physical configuration of the fissionable material at all times in a process as designed must be known, and ways in which the configuration may become altered by off-standard conditions must be predicted. General problems cannot always be solved, and particular situations must be considered individually. The last part of the nuclear-safety problem is administrative or supervisory. In considering the possible events which could lead to a dangerous configuration of fissionable material, it is necessary

to know the rules under which the process is operated and what violations of these rules, intentional or otherwise, may be incurred by operating personnel.

In practice the problem is approached by a detailed study of the process and the equipment, the application of experimentally determined critical parameters to set limits on quantities of material and equipment dimensions, and the formulation of the necessary regulations. Adequate safety factors must be incorporated to cover uncertainties in all parts of the problem.

FACTORS DETERMINING CRITICALITY

An assembly must contain at least the critical mass of fissionable material if it is to support a nuclear chain reaction. This mass is not single-valued but depends very strongly upon a number of conditions.

Neutron Leakage. One condition of importance in determining the critical mass is the leakage of neutrons which might otherwise cause fissions. Neutron leakage depends on vessel shape and on the neutron-reflecting properties of surrounding materials. For example, it is possible to specify vessel dimensions, such as pipe diameters, that give a sufficiently unfavorable area-to-volume ratio to prevent a chain reaction regardless of the quantity of material contained or the pipe length. If the pipe is encased in a cooling jacket or is near other process equipment or structural materials, this dimension will be less than it would be if no neutron reflector were near.

Moderation. Since the fission cross sections of the common fissionable materials increase markedly with decreasing neutron energy, another condition to which the critical mass is very sensitive is the presence of nuclei which effectively slow or moderate neutrons. For example, the critical mass of a fissionable salt may decrease by a factor of 100 as the water content is increased. But upon extensive dilution the critical mass increases without limit because of the neutron absorption by hydrogen so that safety, in the limit, is assured by the chemical concentration alone. The recommendations given below apply to homogeneous and uniform distributions of the fissionable atoms in the moderator.

Density. A third determining factor is the density of the fissionable element including isotopic purity as a special case. The critical mass of the material varies inversely as its density in a manner depending upon other characteristics of the accumulation; it depends in a somewhat similar manner upon the isotopic concentration of the fissionable element.

INSTRUMENTATION

Radiation-detecting instrumentation is not useful in indicating margins of safety in operations except, possibly, in a few particular instances. Any approach to a critical condition is manifested by the multiplication of the ambient neutron field by the fissionable nuclei; so some supply of neutrons is necessary in order to detect the multiplying medium. In some very special cases the material itself is an adequate source of neutrons, which may originate from spontaneous fissions or from nuclear reactions between the constituents of some process materials, to establish an activity level which may be satisfactorily detected. In most instances, however,

the neutron background is too low to serve as a source, and, even though the level increases as more fissionable material is added to the system, it usually does not reach a significant value until the system becomes supercritical. Then the time rate of change of radiation level increases rapidly. To have observed the changes in the subcritical neutron multiplication would have been practically impossible in most instances because of the low initial level and because it is the rate of change in this level that is indicative of the approach to criticality. A possible solution to this difficulty is the inclusion of a strong neutron source in the process system and the observation of changes in the level as material is added. This is the way critical experiments are performed; and experience has shown that the neutron source, the detector, and the fissioning material must be carefully located with respect to each other in order to achieve results which yield meaningful values of the neutron multiplication. To equip process operations in the necessary elaborate manner is generally not practical. Instrumentation has, however, been installed in many operations to indicate the radiation hazard which would exist after an accident has occurred, and reference is made to standard health-physics procedures for a description of recommended equipment. The utility of other than very specially installed detectors can be summarized by saying they are important after an accident, not in predicting that one is imminent.

DESIGN CRITERIA

The bases for the design of equipment and processes for the fissionable isotopes are almost entirely predicated upon results from necessarily restricted critical experiments or upon interpolations or short extrapolations of these results. Many experiments have also been performed which showed that particular situations were not critical—important results but of limited application. In spite of an impressive accumulation of background data, many gaps exist which must be covered by extremely conservative estimates. Thus, the recommendations given here are, in some cases, probably overly conservative. Further, in practice, it has been customary to assume operating conditions to be more severe than they probably will be. Most piping, for example, has in the past been designed on the assumption that it may become surrounded by a thick layer of water—perhaps it will because of the rupture of a water main and the stoppage of drains—but a more important reason for such conservative designs is the unknown neutron-reflecting properties of nearby concrete walls, floors, neighboring water lines and process vessels, and of personnel. The recommendations presented below for partial reflectors are applicable to volumes of fissionable materials surrounded by thin reflectors if the user can assure to his satisfaction that the stated conditions will not be violated. As more confidence is gained, not only in the bases for nuclear safety, but in the predictability of operating conditions, more liberal approaches to the problems will evolve.

RECOMMENDED CONTROLS

Safety Factors. Some factors of safety are incorporated in the recommendations given in this section for the control of nuclear chain reactions. These factors are necessary for at least two reasons: there are, as stated above, uncertainties in the

fundamental information upon which these recommendations are based, and secondly, there is the ever-present possibility of underestimating the fissionable material content of process materials. The latter condition may come about by, for example, poor sampling methods, analytical errors, or the inadvertent combination of two or more batches. There is therefore a safety factor of about 2 in the recommended masses and chemical concentrations. The factor in the values of vessel capacities and dimensions is somewhat less, about 1.3 and 1.1 respectively, since operational errors will not impair safety in these instances, and design and construction errors are less likely than operational ones.

Masses, Capacities, and Dimensions. In Table 15-1 are listed recommended

Table 15-1. Recommended Maximum Parameters of Individual Units *

	Metal (unmoderated)			Aqueous solutions				
	Mass, kg	Cylin- der diam, in.	Slab thickness, in.	Mass, kg	Capacity, liters	Cylin- der diam,† in.	Slab thickness,† in.	Concen- tration,‡ g./liter
Complete Water Reflector								
U-233	3.0	1.5	0.2	0.25	2.0	3.7	0.5	5.5
U-235	11.0	2.5	0.6	0.35	4.8	5.0	1.4	5.5
Pu-239	2.6	1.4	0.2	0.25	3.3	4.5	1.5	3.9
Partial Water Reflector (reflector = 1 in. of water equivalent)								
U-233	4.1	1.9	0.5	0.33	3.0	4.7	1.7	
U-235	15.0	3.0	1.1	0.43	6.0	5.8	2.4	
Pu-239	3.3	1.7	0.5	0.32	5.0	5.7	2.6	

* See Fig. 15-2 for maximum densities to which these limits are applicable.

† Inside dimension of containing vessel.

‡ These values refer to the masses of the fissionable isotope, for example, 5.5 g U-233 per liter of aqueous solution.

maximum values of masses and of vessel capacities and dimensions for three fissionable isotopes as metal and in aqueous solutions. (The recommendations for metal apply to either single units of maximum density or tightly packed components among which other substances cannot be dispersed.) These considerations assume the presence of only those materials usually found in processing operations, such as light water, concrete, graphite, and stainless steel, and assume also that any dispersion of them within the fissionable material is homogeneous. Each limitation will be effective in assuring safety even when all other parameters are optimized, provided that no special reactor materials, such as beryllium and heavy water, are present. The diameter of the cylinders and the thickness of the slabs describe vessels having their other dimensions and contents unlimited.

Given also in the table are recommended maximum chemical concentrations of aqueous solutions of the materials, expressed as mass of fissionable isotope per unit volume of solution. The safety of a process designed on these criteria depends upon the absorption of neutrons by hydrogen; so it is important to note that reference is made to homogeneous aqueous solutions, not to mixtures with other materials where the fissionable elements may have these densities.

In instances where the absence of an effectively infinite water reflector can be assured, the recommendations referred to above can be relaxed. The values given in the lower section of Table 15-1 are applicable to individual units surrounded by a reflector equivalent in effect to a layer of water *no more than 1 in. thick*. The cooling jacket around a cylinder and a thick wall of a steel pipe are examples of such partial reflectors since, for *solutions*, steel, graphite, and water in 1-in.-thick layers are comparable as reflectors. With fissionable *metal*, 1.5 in. of steel or graphite is equivalent in effect to a 1-in.-thick water reflector. The parameters for totally unreflected units are, of course, still larger but such conditions are not usually found in practice since structural members, concrete walls and floors, and even personnel provide some neutron reflection.

Vessel Shape. The masses and capacities recommended in Table 15-1 apply to the fissionable materials in the geometric form requiring the least amount of

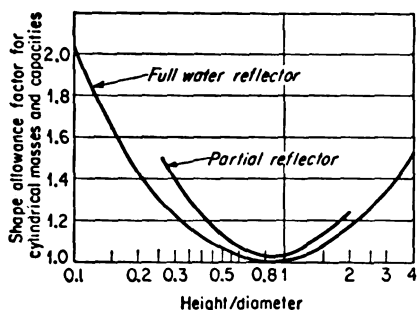


FIG. 15-1. Shape-allowance factors for cylinders.

material to be critical. These values in Table 15-1 can be increased if the shape of the units deviates from the optimum one, considered here to be spherical. Figure 15-1 is a plot of such a shape-allowance factor for cylinders as a function of the height-to-diameter ratio. The ordinates are factors by which the masses and capacities of Table 15-1 can be increased for particular elongated or squat cylinders with full or with partial water reflector. Although the limiting value of this factor is infinity, applicable to the content of the "safe" cylinders

and slabs described in Table 15-1 in which safety is ensured solely by the dimensions, the uncertainty in the function precludes intermediate values beyond those given in Fig. 15-1.

Moderation. The basic criteria given in Table 15-1 apply to aqueous solutions having hydrogen contents or hydrogen atom to fissionable atom ratios giving minimum values of the parameters. (The latter designation is the usual way of expressing concentrations in nuclear-reactor discussions.) It is pointed out that, with isotopically pure fissionable elements, the chemical concentration at which the critical mass is a minimum is significantly different from the concentration for minimum critical volume, the hydrogen to fissionable atom ratio being about 450 for minimum mass and about 45 for minimum volume. If the operating conditions are such that the concentration is known to be different from that giving the minimum value of the controlling parameter, the limitations imposed in Table 15-1 may be relaxed. In Table 15-2 are listed some factors by which the values of masses,

Table 15-2. Allowable Multiplying Factors for Recommended Solution Parameters at Various Chemical Concentrations

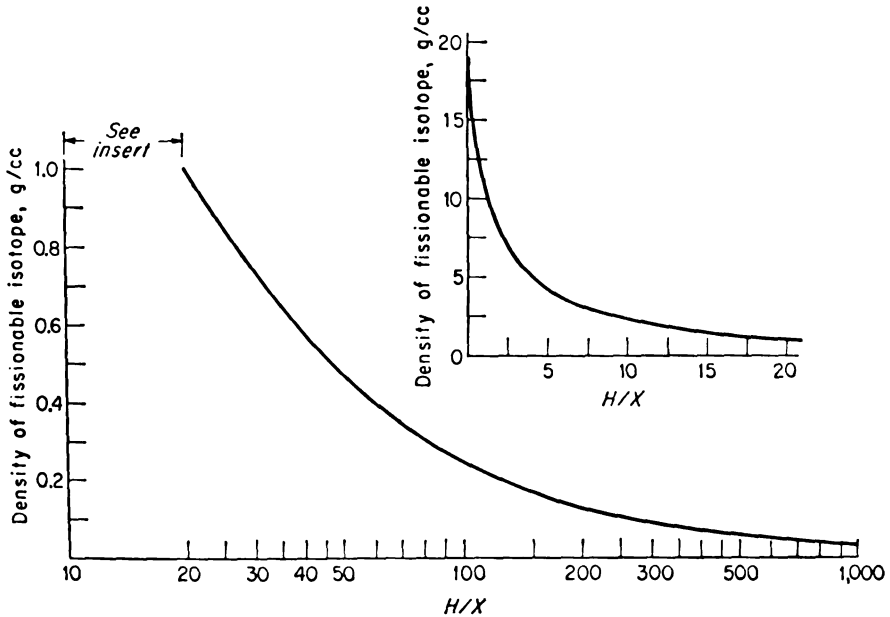
(Concentrations are expressed as Hydrogen atom to fissionable atom ratio H/X)

	Multiplying factor			
	Mass	Capacity	Cylinder diam	Slab thickness
H/X unrestricted	1	1	1	1
H/X less than 20 *	5	1	1	1
H/X less than 100	2	1	1	1
H/X greater than 400	1	2	1.3	1.4
H/X greater than 800	1	3	1.5	2

* Slurries, salts with water of crystallization, and hydrogenous compounds constitute this moderation range.

capacities, and dimensions in Table 15-1 may be multiplied when the solution concentrations lie in the stated ranges.

Density. Figure 15-2 gives the relation between the density of the fissionable isotope and the quantity of intermixed hydrogen expressed as the ratio of hydrogen to fissionable atoms (H/X). All the limits recommended above apply to condi-

**FIG. 15-2. Density of U-233, U-235, and Pu-239 as function of hydrogen concentration H/X.**

tions where the densities do not exceed the values given by this relation. If the density does exceed the relation by the ratio n , the corresponding mass should be reduced by n^2 and the volume limit should be reduced by n^3 . If the density is less than that defined by the relation in Fig. 15-2, the mass or volume should not be increased.

The critical mass of metal as particulates undiluted with other elements, a loose powder, for example, is greater than that of bulk metal because of the density difference. Curve A of Fig. 15-3 gives some allowance factors applicable to the metal

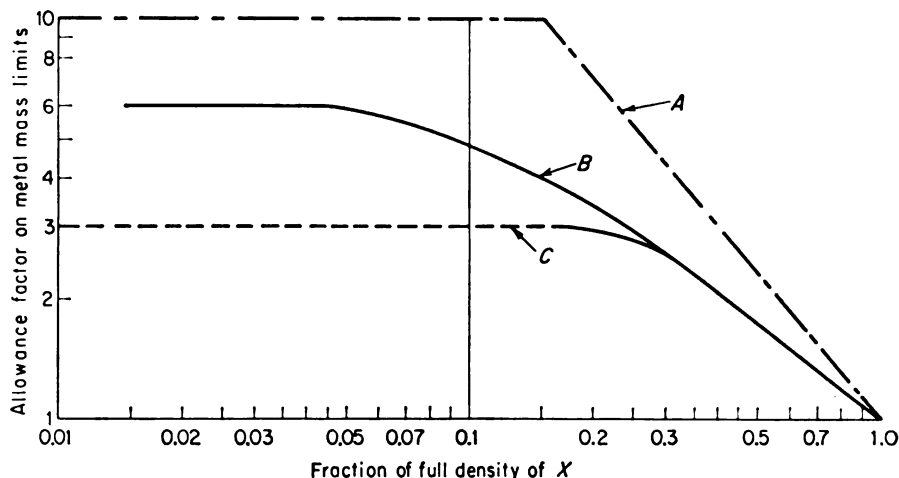


FIG. 15-3. Allowance factor for metal-mass limits for U-233, U-235, and Pu-239 of reduced density. Not applicable to mixtures containing H, D, or Be. Curve A: metal at reduced density. Curve B: compounds and mixtures with elements of $11 \leq Z \leq 83$ (Na to Bi). Curve C: compounds and mixtures containing C, N, O, and F with maximum of eight atoms total per atom of X. Full-density U-233 18.3 g/cm³. Full-density U-235 18.8 g/cm³. Full-density Pu-239 19.6 g/cm³.

masses of Table 15-1 to set mass limits of metal at reduced density. Note that curve A refers to fissionable materials in elemental form with no intermixed foreign material. Curves B and C of Fig. 15-3 refer to compounds and homogeneous mixtures of the fissionable elements with other elements and give, also, allowance factors to be applied to the metal limits of Table 15-1. *These factors do not apply to compounds or mixtures with hydrogen, deuterium, or beryllium.* Curve B gives the factors to be used when the foreign element has an atomic number between 11 and 83 (sodium to bismuth); curve C refers to mixtures and compounds with the lighter elements—carbon, oxygen, nitrogen, and fluorine—in which there is at least one atom of fissionable metal to seven of the lighter atoms. Examples are UF₄, U₃O₈, UC₂, etc.

Isotopic Purity. Critical masses are increased by homogeneously dispersed isotopic contaminants. The most common of these contaminants are Pu-240 in Pu-239 and U-238 in U-235. The quantities of Pu-240 usually encountered are too small to be considered quantitatively and merely increase the margin of safety in the recommended limits. The range of U-235:U-238 concentrations is, of course, very wide,

and the available information on which designs can be based is limited. The extent to which the U-235 unmoderated-metal mass limits of Table 15-1 can be increased when appreciable quantities of U-238 are present is shown in Fig. 15-4. It is emphasized that these factors apply *only* to pure *uranium metal* in the complete absence of any interspersed material. If the metal density is less than the full value, an additional factor based on the uranium (not U-235) density from curve A of Fig. 15-3 may be applied.

For solutions of salts of uranium the values of limiting masses, capacities, and cylinder diameters given in Table 15-3 at several concentrations of U-235 in uranium have been calculated.* They apply to any H:U-235 atomic ratio and full-water reflector provided the total uranium density does not exceed the values of U-235 densities given in Fig. 15-2. The recommendations probably contain safety factors as large as those applying to the corresponding entries in Table 15-1.

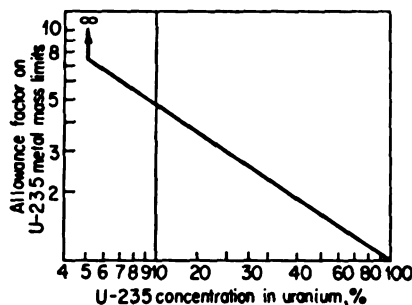


FIG. 15-4. Allowance factor for U-235 metal-mass limits for U-235 diluted with U-238. Not applicable to solutions, mixtures, or lattices of metal in liquids.

Table 15-3. Recommended Limiting Parameters for Partially Enriched Uranium

(Applicable to any H:U-235, fully water-reflected provided density does not exceed values in Fig. 15-2)

U-235 content of uranium, %	Mass U-235, kg	Capacity, liters	Inside cylinder diam, in.
40	0.41	6.7	6.0
20	0.48	9.5	6.9
10	0.60	14.0	8.2
5	0.80	27.0	10.2
2	2.00	95.1	16.0
0.8	36.00	3,900	58.0
≤0.7	∞	∞	∞

Neutron Absorbers. No extensive use of materials of high-neutron-absorption cross section is recommended as primary safety controls. They must be homogeneously mixed with fissionable material to be effective internal poisons and present, therefore, subsequent decontamination problems. Surrounding a lightly reflected vessel with cadmium or boron may, in fact, actually decrease its safety, because any additional adjacent mass will scatter back into the system some neutrons which

* The preparation of these recommendations by H. F. Henry *et al.* of the Oak Ridge Gaseous Diffusion Plant is gratefully acknowledged. The values for 5 per cent U-235 content are based on experiment.

would otherwise leak away. Placing one of these absorbers between a container and a source of low-energy neutrons, another vessel filled with a solution of fissionable material, for example, can increase the safety, although there are no generally applicable rules. Nitrogen in the nitrate solutions often used in chemical processes and Pu-240, present as an impurity in Pu-239, have been considered as increasing the margin of safety rather than of prime importance in establishing design criteria.

Pipe Intersections. The limits of cylinder diameters recommended in Table 15-1 apply to individual units and must be reduced when connections of pipes are considered. Table 15-4 gives maximum inside diameters of piping joined by 90°

Table 15-4. Recommended Diameters in Inches of 90° Intersecting Pipes Containing Fissionable Aqueous Solutions

($H/X \geq 20$; density not exceeding that of Fig. 15-2)

	U-233	U-235	Pu-239
Tees:			
Full-water reflector	2.6	3.5	3.2
Partial reflector	3.3	4.1	4.0
Crosses:			
Full-water reflector	2.1	2.9	2.6
Partial reflector	2.7	3.3	3.3

connections and containing fissionable solutions at $H/X > 20$ with densities not exceeding those of Fig. 15-2.

Multiple-unit Arrays. Of importance in component layout, either in plant design or in shipment and storage arrays, are the spacing requirements among single units which are individually subcritical. The exchange of leakage neutrons among components of an array is measurable when they are separate in air by the order of 10 ft, in contrast to the almost complete attenuation of neutrons by a 12-in.-thick layer of water. In the absence of complete and generally applicable rules for component spacing it has been the practice, for a limited number of units, to size the individual components to be safe even when surrounded by water and to separate them at least 1 ft side to side. In this arrangement, the units are effectively isolated by the water when flooded and, in the absence of the water, are sufficiently subcritical to make the exchange of leakage neutrons among them insignificant.

General References

Paxton, H. C., and Glen Graves, Critical Masses of Fissionable Metals as Basic Nuclear Safety Data, *Los Alamos Sci. Lab. Rept.*, LA-1958 (deleted), April, 1956. (30¢)

Callihan, Dixon, J. W. Morfitt, and J. T. Thomas, Small Thermal Homogeneous Critical Assemblies, *Proc. Intern. Conf. Peaceful Uses Atomic Energy, Geneva*, vol. 5, p. 145, 1956.

Horton, C. C., and J. D. McCullen, Plutonium-Water Critical Assemblies, *Proc. Intern. Conf. Peaceful Uses Atomic Energy, Geneva*, vol. 5, p. 156, 1956.

Callihan, Dixon, Nuclear Safety in Processing Reactor Fuel Solutions, *Nucleonics*, vol. 14, no. 7, p. 39, July, 1956.

Thomas, J. T., J. K. Fox, and Dixon Callihan, A Direct Comparison of Some Nuclear Properties of U-233 and U-235, *Nuclear Sci. Eng.*, vol. 1, p. 20-32, 1956.

Henry, H. F., A. J. Mallett, and C. E. Newlon, *Rept. K-1019*, Fourth Revision, May, 1958. (\$1.50)

Callihan, Dixon, W. J. Ozeroff, H. C. Paxton, and C. L. Schuske, *Rept. TID-7016*. (\$1.00)

Brown, J. R., B. H. Noordhoff, and W. O. Bateson, *Rept. WAPD-128*, May, 1955. (35¢)

Kruesi, F. E., J. O. Erkman, and D. D. Lanning, *Rept. HW-24514*, May, 1952.

Unclassified U.S. Atomic Energy Commission reports may be purchased from the Office of Technical Services, Department of Commerce, Washington 25, D.C.

Section 16

RADIATION HYGIENE CHEMISTRY

By

MORRIS F. MILLIGAN, Ph.D., *Los Alamos Scientific Laboratory, Los Alamos,
New Mexico.*

Strontium-90.....	16-5
Tritium.....	16-7
Actinium.....	16-8
Americium.....	16-9
Protactinium.....	16-9
Uranium (Radiochemical).....	16-10
Radium.....	16-12
Uranium (Fluorimetric).....	16-14
Plutonium.....	16-16
Thorium.....	16-19
Polonium.....	16-22
Ruthenium.....	16-24

RADIATION HYGIENE CHEMISTRY

Morris F. Milligan

Radiation hygiene chemistry (commonly called health radiochemistry, occupational biochemistry, or bio-assay) may be informally defined as the chemistry involved in measuring, or preparing to measure, toxic radioactive nuclides in samples of almost any origin, including filter media, soil, water, swipes from surfaces suspected of contamination, or biological materials. Most of the procedures given in this chapter involve the analysis of human urine, for three reasons: such procedures are readily adapted or simplified to accommodate samples of other types; the great majority of samples processed in laboratories doing this work are of human urine; and such analyses give the most direct information about the body content of, and hence about the exposure to, radioactive materials.

Rigorous **special laboratory requirements** must be met to satisfy the demand for sensitive and accurate analyses; ideally, the laboratory should be isolated from research or production facilities; it should have an independent supply of filtered washed air; it should be ample enough in size that each type of procedure may have its own area, minimizing the possibility of cross-contamination; unknown samples suspected of containing considerable radioactivity should be analyzed in a separate area; stock solutions containing radioactivity (for use as spikes or standards) should not be kept within the laboratory proper; the laboratory hoods, lined with acidproof material, should be exhausted at a minimal average face velocity of 100 ft/min; and in the event of fortuitous contamination, the laboratory must be designed so that decontamination procedures will not be hindered. The latter requirement is at least partially met by providing working surfaces of stainless steel fabricated without sharp inside corners. It is to be emphasized that enumerated above are ideal conditions, and that acceptable radiation hygiene chemistry has been carried out under much less favorable circumstances.

The **counting-room location** is of considerable importance; for convenience, it should be in the same building, but it should certainly not be adjacent to rooms in which chemical manipulations are made. The counting room should be adequately shielded from occasional fluctuations in atmospheric background; it should be protected against electrical interference of all types; it should be furnished with a reasonable supply of air at constant temperature and humidity; and it must be adjoined by a suitable technical facility for the maintenance of counting equipment. If nuclear film techniques are to be used for the assessment of very low orders of activity (<0.1 disintegration/min), suitable provision for dark-room work and microscopy obviously must be provided; precautions against the advent of contamination are also of importance here.

The **selection of the apparatus used in counting** is based upon consideration of the type of radiation being counted (alpha, beta, or gamma), the energy of the radiation, the sensitivity desired, and occasionally the possibility of internal contamination of the counting device. Since the samples may contain vanishingly small but highly significant amounts of the desired constituent, counting equipment must be designed for continuous low-level operation at low backgrounds and constant efficiencies.

Alpha counting is carried out on thin films of the alpha-emitting material supported by disks, plates, planchets, or dishes of stainless steel, platinum, silver, aluminum, or copper. The thin films must have a low-density thickness ($<1 \text{ mg/cm}^2$) because attenuation of the short-range alphas by self-absorption becomes troublesome at higher values. It is thus necessary to isolate the desired constituent in a preparation of reasonably low mass; most of the chemical procedures described later are designed to accomplish this end. Alpha counting is customarily carried out on one of two types of instruments, proportional gas-flow counters or scintillation counters. Gas-flow counters operate at 50 per cent efficiency with backgrounds consistently below 4 counts/hr; they are thus eminently suitable for low-level work with carefully prepared samples; unfortunately they are susceptible to serious internal contamination, although modern chamber design has alleviated to some extent the problem of decontamination. Scintillation counters are considered more useful in the counting of less carefully prepared samples, and in the counting of filter paper, in which form particulate air samples are received, since the sensitive volume of the detector cannot be contaminated; further, scintillation counters can be readily provided with suitable discriminator circuits which will tend to reject alphas of other than the desired energy. The monoenergetic nature of alpha particles makes possible the positive identification of the radioactive species present; for this measurement, it is necessary to resort to pulse-height analysis or alpha spectrometry, for which facilities are not always available; only samples with considerable activity can be examined by these means. The use of nuclear films, although uneconomical of time, permits the measurements of much lower levels of alpha activity than by any other means. In this technique, a carrier-free (usually electrodeposited) preparation is allowed to stand in contact with an alpha-sensitive film for a definite period (about 1 week); the film is examined microscopically for characteristic tracks after development (it is obviously desirable that the sample cover as small an area on the plate as possible); comparison of the unknown with standards affords a calculation of the activity present.

The apparatus used in **beta counting** is dependent upon the energy of the radiation under measurement. For most preparations, a thin ($1 \text{ to } 2 \text{ mg/cm}^2$) mica-end-window Geiger-Müller counting tube will suffice. In low-level beta counting, recourse must be had to the use of efficiently shielded detectors protected with a multiplicity of other tubes arranged in anticoincidence; by such means, backgrounds of 1 count/min may be achieved, particularly if screen-wall counters or thin-wall (Mylar) counting detectors are used. For the measurement of very weak beta radiation, such as that from tritium, it is necessary to do internal counting, either using the active material in the gaseous phase as partial filling for counter tubes or ionization chambers, or counting in a liquid-scintillation system.

In the **counting of gamma emitters**, either of two general types of apparatus may

be used; high-pressure ionization chambers are useful for measuring activity in liquid samples with greater efficiency than possible with end-window tubes, but the greatest efficiency is attained with 4 pi liquid-scintillation systems; in this case, the sample (usually liquid in a polyethylene container) is virtually surrounded by an annulus of liquid scintillator in which are placed a number of photomultiplier tubes, the whole apparatus being efficiently shielded. Gamma counting may be carried out with little or no chemical preparation unless interfering elements are present.

The actual **chemical procedures** selected for use in effecting separations of active material from the accompanying mass of the sample must be chosen on the basis of several criteria: speed, ease of operation, sensitivity, specificity, and economy (not necessarily in order of importance). Since it is seldom possible for a given operation to fulfill these requirements simultaneously, compromises are usually effected; perusal of the specific methods given later will reveal instances of such compromises. In general, there are five methods of chemical separation used in the separation of radioactive materials from samples as received or after suitable pretreatment (e.g., ashing): coprecipitation, ion exchange, volatilization, electrodeposition, and solvent extraction.

Coprecipitation is the method of separation utilizing the tendency of a precipitate formed in solution to carry other ions with it. By a judicious choice of precipitating ions, it is possible to carry down a preponderance of the ions of a given trace element. If nonradioactive isotopes of the desired constituent are available, they are obviously the ions of choice; very often no such ion is available and it is necessary to select an ion somewhat similar in size and charge to the one desired (nonisotopic carrier). For example, radium may be carried from a solution in which a precipitate of barium sulfate is formed; radium is similarly carried by lead chloride; plutonium is carried by either lanthanum fluoride or bismuth phosphate. The precipitates formed in this technique are customarily harvested by centrifugation rather than filtration. Scavenging denotes the coprecipitation of interfering ions leaving the desired constituent in solution.

The **use of ion-exchange resins** (either anionic or cationic) makes possible (at least theoretically) the separation of any ion or group of ions from large masses of accompanying material. The factor making this possible is the tendency of ion-exchange materials to combine chemically with ions of various sizes and charges with varying degrees of affinity. Thus, one may place a mixture of ions on an ion-exchange resin (either in a column or by batch operation) and remove one at a time by a suitable selection of solvents; alternatively, conditions may be chosen so that a given ion will remain in solution and be thus separated from ions combined with the resin. An example of this type of operation is given in the determination of barium and strontium in urine, whereby selection of solvents and complexing agents makes it possible to separate these ions from all others.

Volatilization is a separation technique used in radiation hygiene chemistry only for those elements which can readily be converted to the gaseous state; tritium is customarily converted to a suitable hydrocarbon or to the element, while carbon may be caused to form carbon dioxide. Special counting techniques are obviously indicated if the material is to be counted in the gas phase.

There are two types of **electrodeposition** differing widely in fundamental action: (1) spontaneous electrochemical exchange, whereby polonium, for example, is

separated on a nickel, copper, or silver surface, and (2) electroplating proper, whereby plutonium or uranium may be deposited on appropriate base metals as a final step in preparation for counting; impressed electrodeposition is not so valuable as a means of separation as it is for the preparation of very thin uniform films of the desired constituent.

Solvent extraction is an efficient and useful means of chemical separation, depending upon the distribution coefficient of the desired constituent (often as a complex) between the aqueous phase (admixed with extraneous material) and an insoluble organic phase in which the desired constituent is to be selectively extracted. An example of solvent extraction is the separation of plutonium from acid solution with a benzene (or toluene) solution of TTA (thenoyltrifluoroacetone) or the separation of uranium from nitric acid solutions with ether, the latter being often carried out in an apparatus for continuous extraction.

STRONTIUM-90

Strontium-90 is the fission product of greatest interest to radiation hygienists because of its high fission yield, its long half-life (28 years), and its chemical similarity to the biologically important calcium.

The **separation of strontium** may be brought about by precipitation as phosphate, as carbonate, or as oxalate. It may be isolated as nitrate in fuming nitric acid, or by ion exchange. Since the Strontium-90 beta particle is rather weak (0.54 Mev), it is advantageous to age the isolated strontium and then "milk" from it the only radioactive daughter, Yttrium-90, which emits a 2.2-Mev beta particle; equilibrium is 97 per cent complete in 2 weeks and 99 per cent complete in 3 weeks. Yttrium isolation is accomplished by precipitation as hydroxide, as phosphate, by extraction with TTA (thenoyltrifluoroacetone) in benzene, or by ion exchange. The isolated yttrium may be identified by its characteristic half-life (64.6 hr) and is counted on conventional equipment, or on low-background counters to achieve greater sensitivity.

The **determination of Strontium-90 in urine** is done most simply by coprecipitating as oxalate the strontium with the calcium normally present, and counting this precipitate. Considerably greater accuracy is obtained by isolating carrier-free strontium (and barium) by the following ion-exchange procedure (Farabee, L. B., *Radiochemical Analysis of Strontium and Barium in Human Urine*, *A.M.A. Arch. Ind. Health*, vol. 17, pp. 200-203, 1958):

1. Add concentrated hydrochloric acid to a 1,500-ml urine sample contained in a 2-liter beaker till the urine is 0.1N in acid.
2. Heat the sample to a temperature of 85 to 90°C.
3. Add 6 ml 6M phosphoric acid.
4. With vigorous mechanical stirring, add dropwise 6M sodium hydroxide to a pH of 8 to 10, using pH test paper.
5. Allow the precipitate to settle for 2 hr or longer.
6. Decant and discard the supernatant solution, taking care not to disturb the precipitate.
7. Centrifuge the remaining slurry at 1,500 rpm for 5 mins.
8. Decant and discard the supernatant liquid.

9. Dissolve the precipitate in about 15 ml concentrated nitric acid.
 10. Transfer the solution to the 2-liter beaker in which the original phosphate precipitation was carried out.
 11. Heat the solution and destroy organic matter by adding alternately concentrated nitric acid and 30 per cent hydrogen peroxide.
 12. Repeat the previous step until a white residue remains; then take to dryness.
 13. Allow to cool; wash down the walls of the beaker with about 20 ml distilled water; add 2 to 3 ml concentrated hydrochloric acid and heat until solution is complete. Dilute to 800 ml with distilled water.
 14. Adjust the pH of the solution to about 9.0 with 1*N* sodium hydroxide.
 15. Add 2 ml Eriochrome Black T solution (prepared by dissolving 0.5 g Eriochrome Black T and 4.5 g hydroxylamine hydrochloride in 100 ml ethanol and filtering).
 16. Add a solution of 7.5 per cent versene until the indicator changes from wine-red to blue at pH 10.5. The versene solution (7.5 per cent) is prepared by dissolving 75 g technical-grade tetrasodium ethylenediaminetetracetate in about 800 ml distilled water, filtering, and diluting to 1 liter with distilled water.
 17. Reduce the pH to 5.5 with concentrated hydrochloric acid, using 1*N* hydrochloric acid as the desired pH is approached.
 18. Pass the solution over the resin column at a flow rate not greater than 8 ml/min/cm². The resin column is Dowex 50 × 12 resin (50 to 100 mesh) in the sodium form contained in a pyrex tube 18.5 cm long and 1.8 cm inside diameter.
 19. Rinse the beaker and walls of the filling funnel with about 50 ml distilled water. Discard the effluents.
 20. Pass 800 ml citrate-versenate solution over the resin column at a flow rate of 4 ± 0.8 ml/min/cm². The citrate-versenate solution is prepared by dissolving 10.93 g citric acid monohydrate in distilled water, adding 100 ml 7.5 per cent versene solution, diluting the mixture to 1 liter, and adjusting to pH 5.0 with 6*M* sodium hydroxide.
 21. Discard the effluents.
 22. Pass 800 ml 0.5*N* hydrochloric acid over the resin column at a flow rate of 4 ± 0.8 ml/min/cm².
 23. Discard the effluents.
 24. Pass 200 ml 6*N* nitric acid over the resin column at a flow rate of 2 ml/min/cm², collecting the eluate in a 400-ml beaker.
 25. Evaporate the eluate almost to dryness.
 26. Wash the contents into a 50-ml beaker with distilled water. Rinse the large beaker with nitric acid and water, adding the rinsings to the smaller beaker.
 27. Evaporate the liquid in the small beaker to dryness.
 28. Dissolve the residue in about ½ ml 1*N* nitric acid. Using a transfer pipette, transfer this liquid to a stainless-steel counting dish. Wash the walls of the beaker with another ½-ml nitric acid and transfer this to the dish.
 29. Dry the sample under infrared heat.
 30. Beta count on a conventional end-window G-M counter.
- Should it be desired to "milk" the parent strontium preparation for Yttrium-90, complete separation of calcium and strontium is not invariably necessary. The

following **extraction procedure with TTA** is useful (Goldin, A. S., Determination of Strontium-90, *USAEC TID-7517*, part 1b, pp. 323-334, 1956):

1. The isolated aged strontium preparation is brought to about 20 ml with ammonium acetate-acetic acid buffer at pH 5, and extracted three times with 5-ml portions 5 per cent TTA in benzene. The time of the last extraction is noted as the end of the growth period and start of the decay period.

2. Wash the separatory funnel several times with water, return the combined benzene extracts to it, and wash three times with 5-ml portions ammonium acetate-acetic acid buffer (pH 5). Transfer the benzene layer to a clean separatory funnel; wash the first funnel with 5 ml benzene, and add to the benzene extract.

3. Extract the benzene layer three times with 5-ml portions 1*N* nitric acid, running the aqueous layer into a stainless-steel counting dish. Evaporate to dryness and count.

In selecting an isolation procedure, consideration should be given to the probable age of the fission products in the sample. Barium-140 should be separated, usually by barium chromate scavenging, if the fission products are known to be less than one year old; for the most accurate work, barium chromate scavenging is invariably included in the procedure.

Excellent and detailed procedures are available for the **determination of Strontium-90** in a wide variety of materials. [Bryant, F. J., Chamberlain, A. C., Morgan, A., and Spicer, G. S., Radiostrontium Fallout in Biological Materials in Britain, *UKAERE HP/R 2-56*, amended, 1957. Whitney, I. B. (ed.), Manual of Standard Procedures, Analytical Branch, Health and Safety Laboratory, New York Operations Office, *USAEC NYOO-4700*, 1957.]

TRITIUM

The **radiochemical analysis for tritium** is complicated because of the very low beta-particle energy, the maximum energy being about 18 kev. It is thus necessary to measure tritium either as a tritium-containing gas in a suitable counting tube or ionization chamber, or as a component of a suitable liquid-scintillation system. Since tritium entering the body rapidly equilibrates with the body water, it is necessary only to determine the tritium oxide concentration in the body water in order to evaluate the amount of tritium in the body.

Techniques involving the **conversion of tritium oxide present in body fluids** to the gaseous state involve the use of vacuum manifolds, special ionization chambers or counting tubes, and considerable electronic equipment. Tritium oxide may be converted to the element by the action of hot zinc, or by the action of metallic calcium; it may be converted to acetylene by the action of calcium carbide, to methane by aluminum carbide, or to any desired hydrocarbon by a suitable Grignard reagent (RMgX). Any of these gaseous compounds may be used to fill suitable ionization chambers for ion-current measurements with an electrometer; alternatively, they may be loaded into suitable tubes for counting in the proportional or Geiger regions. If the counting tube is filled with hydrogen, an accessory quenching circuit must be included in the electronics; if, however, argon and ethylene are mixed with the hydrogen so that the mixture is self-quenching, more conventional circuitry is used.

Liquid-scintillation systems offer the most effective means of measuring tritium in solution; for this analysis, a fast-coincidence liquid-scintillation counter must be used. Analogues of the original instrument are now available commercially (Technical Measurements Corporation, New Haven, Conn.; Packard Instrument Company, LaGrange, Ill.).

The following **liquid-scintillation method** is useful for the routine analysis of urines for tritium (Okita, G. T., Spratt, J., and Leroy, G. V., *Liquid Scintillation Counting for Assay of Tritium in Urine*, *Nucleonics*, vol. 14, no. 3, pp. 76-79, 1956):

1. Decolorize 30 ml urine with 1 g activated charcoal and filter.
2. Place 2.0 ml decolorized urine, 0.1 ml distilled water, and 20.0 ml absolute ethanol in a weighing bottle 50 by 60 mm.
3. Add 28.0 ml toluene and 100 mg 2,5-diphenyloxazole. Mix thoroughly.
4. Chill to a temperature of 6°C.
5. Count.

A somewhat similar **liquid-scintillation method** using a slightly different liquid scintillator is as follows (Langham, W. H., Eversole, W. G., Hayes, F. N., and Trujillo, T. T., *Assay of Tritium Activity in Body Fluids Using a Liquid Scintillation System*, *J. Lab. Clin. Med.*, vol. 47, pp. 819-825, 1956):

1. Prepare the scintillation solution as follows: 7 g 2,5-diphenyloxazole (PPO), 50 g naphthalene, and 50 mg 1,4-di-(2-(5-phenyloxazolyl))-benzene (POPOP) is made to 1 liter with *p*-dioxane.
2. Two milliliters urine is filtered with suction through 0.5 g activated charcoal in a 2-ml fine-pore fritted-glass funnel.
3. One milliliter decolorized urine is transferred to a 40-ml centrifuge tube and 24 ml scintillation solution is added.
4. The turbid mixture is centrifuged at 2,500 rpm for 10 mins.
5. The clear supernatant fluid is transferred to a counting bottle (Kimble Opticlear 10-dram vial with polyethylene stopper).
6. The sample is chilled at 2 to 3°C for 1 hr.
7. Count.

Neither of the techniques above yields absolute figures, i.e., it is necessary to run blanks and standards concomitantly in order to calculate the amount of tritium present in the sample. For the most accurate results, it is necessary to run internal standards (sample plus added tritium) in order to determine the counting efficiency of the system. For routine work, an average value can be used for counting efficiency; in this situation, only those samples containing significant amounts of tritium would be run each with its own internal standard.

ACTINIUM

The **determination of actinium** in urine is complicated by the very low energy of the beta particle emitted from Actinium-227. Fortunately, there are alpha-emitting daughters of this nuclide, and by deferring the actual counting procedure after isolation of actinium, growth of these daughters will allow successful alpha counting. Figure 16-1 illustrates the growth of these daughters into a mounted sample of Actinium-227.

The radiochemical isolation of actinium may be accomplished by the procedure

given on page 16-16 for plutonium, whereby actinium is carried serially by bismuth phosphate and by lanthanum fluoride. Alternatively, actinium may be carried from solution by cerium phosphate following removal of radium, although the procedure is tentative. The carrying of actinium on lead sulfate from hot acid solution has been applied to the analysis of rat urine and could be applied with appropriate modifications to human urines. (Kirby, H. W., and Brodbeck, R. M., Determination of Radium, Actinium, and Thorium in Human Urine, *USAEC MLM-1003*, 1954. Rogers, N. E., and Watrous, R. M., Separation of Actinium and Its Daughters by Means of Lead Sulfate, *Anal. Chem.*, vol. 27, pp. 2009-2012, 1955.)

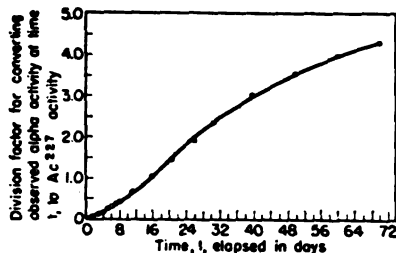


FIG. 16-1. Growth of actinium daughters.

AMERICIUM

The **radiochemical determination of americium** in urine is carried out by a slight variation of the process given on page 16-16 for the determination of plutonium whereby plutonium is carried serially by bismuth phosphate and by lanthanum fluoride. The variation consists in removing traces of iron by extraction with thiocyanate-amyl alcohol-ether before the coprecipitations, and in using cerous fluoride in place of lanthanum fluoride in the final coprecipitations, British practice favoring cerium in place of lanthanum. It is assumed that any actinium, thorium, plutonium, neptunium, or curium in the sample would be included with the americium. (Jenkins, E. N., and Sneddon, G. W., Analysis of Urine for Traces of Americium and Other Alpha Emitters, *UKAERE C/R 1399*, 1954.)

PROTACTINIUM

The **analysis of protactinium** in urine is carried out by ashing the urine with nitric acid and extracting the protactinium with diisopropyl ketone from a hot 10M hydrochloric acid solution of the urine ash; recoveries of 84 per cent are reported. (Russell, E. R., Determination of Protactinium in Urine, *USAEC ANL-4211*, 1948.)

1. Transfer the 24-hr urine sample to a beaker; rinse the original container with 100 ml concentrated nitric acid, add the rinsings to the urine, cover with a watch glass, and evaporate to dryness.
2. Cool; add sufficient concentrated nitric acid to cover the solids; evaporate slowly to dryness.
3. Repeat the previous step until the ash is perfectly white.
4. Add 50 ml concentrated hydrochloric acid and evaporate to near dryness. Add 60 ml 10M hydrochloric acid and heat 15 min with occasional stirring.
5. Allow the insoluble residue to settle (keeping the solution hot); then decant the solution into a 125-ml separatory funnel; wash the beaker and residue with

20 ml hot 10M hydrochloric acid and add the washings to the separatory funnel; ignore any solid material inadvertently transferred to the separatory funnel.

6. Add 25 ml diisopropyl ketone to the separatory funnel and shake 10 min. Discard the aqueous layer and collect the ketone in a 50-ml beaker. Wash the funnel with 5 ml ketone and add washings to the beaker.

7. Evaporate the ketone solution to dryness in an oven at 100 to 110°C. Higher temperatures cause formation of large amounts of nonvolatile organic residue.

8. Cool, cover the solid with concentrated nitric acid and evaporate to dryness; cool, cover the solid with 30 per cent hydrogen peroxide, and evaporate to dryness; repeat, alternating with acid and peroxide until the residue is white.

9. Add sufficient acid to moisten the bottom of the beaker. Warm and transfer to a stainless-steel counting disk, using as much warm acid as necessary.

10. Evaporate to dryness and flame gently.

11. Alpha count.

URANIUM (RADIOCHEMICAL)

Radiochemical analysis for uranium is necessary if one suspects the presence of isotopes of higher specific activity than normal; such a measurement in conjunction with fluorometric analysis can give some idea of the uranium-isotope distribution in a sample. Isolation for radiochemical analysis of uranium is usually accomplished by extraction of ashed urines with diethyl ether or with dibutylphosphoric acid, although uranium has been electrodeposited directly from urine on a silver disk following simple digestion with nitric acid or following ashing and removal of calcium as oxalate. Other techniques not at present used in radiation hygiene chemistry would be useful; these include paper chromatography, ion exchange, acetylacetone extraction, and extraction of uranium at valences of 3 or 4.

Extraction with ether is the basis of the following method for the **isolation of uranium** from urine with recoveries of about 86 per cent (Bouton, R. Z., Method for the Detection of Enriched Uranium in Human Urine, *USAEC KAPL-667*, rev., 1953):

1. To a 24-hr urine sample contained in a 2-liter Erlenmeyer flask add concentrated nitric acid to a distinct darkening in color.

2. Evaporate to dryness overnight at low heat.

3. Boil the residue with 100 ml concentrated nitric acid to dissolve all solids.

4. Transfer the solution to a 250-ml beaker, washing the flask with distilled water and adding the washings to the beaker.

5. Evaporate carefully to dryness.

6. Add 10 ml concentrated nitric acid and evaporate to dryness.

7. Repeat the previous step until a white ash is obtained.

8. Add 60 ml aluminum nitrate solution [2M in $\text{Al}(\text{NO}_3)_3$ and 0.5M in nitric acid]; allow to stand overnight.

9. Stir to aid in solution of the residue; transfer the solution to a 250-ml separatory funnel.

10. Add 30 ml aluminum nitrate solution to the beaker, stir, and transfer the solution to the same funnel.

11. Use two more 10-ml portions of aluminum nitrate solution to dissolve the residue and aid in complete transfer of the sample to the same funnel.

12. Add 50 ml diethyl ether; shake for 20 min, venting occasionally to relieve pressure in the funnel.

13. Allow to stand 20 min for complete phase separation.

14. Discard the aqueous layer.

15. Transfer the ether layer from the top of the funnel into a 100-ml beaker containing 5 ml distilled water; particular care must be used to see that none of the aqueous phase is transferred.

16. Allow the ether to evaporate, transfer the water solution to a stainless-steel planchet or disk, dry at low heat, flame briefly, and alpha count.

Extraction of uranium with dibutylphosphoric acid (DBP) is the basis of the following procedure; recoveries by this means average 85 per cent, and hazards attendant upon the use of ether are obviated (Campbell, E. E., Head, B. M., and Milligan, M. F., Determination of Uranium Alpha Activity in Urine, *USAEC LA-1920*, 1955):

1. Place 100 ml urine and 25 ml concentrated nitric acid in a suitable flask or beaker and evaporate to dryness.

2. Add 10 ml concentrated nitric acid and evaporate to dryness.

3. Repeat the previous step until a white ash is obtained.

4. Dissolve the residue in 10 ml 20 per cent nitric acid by warming.

5. Transfer the solution to a 40-ml centrifuge tube and dilute to 30 ml with distilled water.

6. Allow the solution to cool to room temperature and add 1 ml DBP solution (0.4M in carbon tetrachloride).

7. Stir mechanically for 10 min with the tip of a glass stirring rod just above the solvent layer.

8. Transfer the solvent layer to a platinum dish or planchet with the aid of a transfer pipette equipped with a syringe; evaporate the carbon tetrachloride with gentle heat.

9. Repeat the three preceding steps twice, in each case transferring the solvent layer to the same platinum dish for evaporation.

10. Wash the aqueous phase once with 1 ml carbon tetrachloride, transferring this to the platinum dish.

11. Increase the heat slowly until the material on the dish is charred and appears dry.

12. Flame carefully, gradually increasing the temperature until boiling ceases and removing from the flame as soon as the last black residue has disappeared.

13. Alpha count.

Electrodeposition is the basis of the following short procedure, which is obviously not highly specific but which is of considerable utility in the routine analysis of urines from personnel suspected of exposure to enriched uranium (Hoover, R. L. (ed.), Proceedings of Bioassay and Analytical Chemistry Meeting, *USAEC NLCO-595*, 1955):

1. Twenty milliliters urine and 15 ml concentrated nitric acid in a 150-ml beaker is taken to dryness on the steam bath.

2. The solids are dissolved in about 10 ml distilled water.

3. Add 20 ml ferric ammonium oxalate solution (prepared by dissolving 454 g ammonium oxalate and 3.44 g ferric ammonium sulfate in 16 liters distilled water).

4. Adjust the pH of the solution to 5.0 with ammonium hydroxide.
5. Transfer the solution quantitatively to an electrolysis assembly comprising a glass chimney, a silver disk cathode 1.75 by 0.002 in., a spiral platinum anode suspended in the solution, and accessory fittings.
6. Electrolyze at a current of 2.5 amp until the solution foams or until a temperature of 95°C is attained.
7. Reduce the current to 1.5 amp and continue plating for a total period of 50 min.
8. Remove the plated silver disk; blot dry with absorbent tissue.
9. Alpha count.

RADIUM

The **determination of radium** may be carried out by one of two general methods; either the first daughter of radium, the rare gas radon, may be separated and counted (the emanation method), or radium itself may be separated and alpha counted together with its daughter nuclides. The latter procedure is expeditious and certainly involves the use of less expensive equipment.

Coprecipitation with carriers is most commonly used in the **isolation of radium**; compounds used as carriers include lead sulfate, barium chloride, barium sulfate, lead chromate, barium chromate, and calcium phosphate. Barium nitrate in 80 per cent nitric acid has also served to carry radium.

Increased sensitivity in radium counting may be achieved by deferring the actual counting procedure until growth of daughter alpha activities has taken place.

For Radium-226, build-up of these activities follows the build-up of radon as indicated in Fig. 16-2. At equilibrium (99 per cent complete in 30 days), the total alpha activity is four times the radium alpha activity because of the three alpha-emitting daughters of consequence.

Should the presence of radium isotopes other than Radium-226 be suspected, the preparation should be counted at intervals after separation, and the change in activity should be compared with those calculated for the various isotopes.

Radium-226, as indicated above, increases in total alpha activity to four times the radium alpha activity with a 3.825-day half-life; Radium-223 shortly after separation decays with an 11.2-day half-life, the total alpha activity being four times the radium alpha activity; and Radium-224 total alpha activity increases rapidly by a factor of 3, finally reaching transient equilibrium and decaying with a half-life of 3.64 days.

Serial coprecipitation with lead sulfate and with barium chloride is the basis for the following **method for isolating radium** from 24-hr urine samples; radium recoveries of 98 per cent are obtained (Russell, E. R., Lesko, R. C., and Schubert, J., Direct Determination of Radium in Exposed Individuals, *Nucleonics*, vol. 7, no. 1, pp. 60-64, 1950):

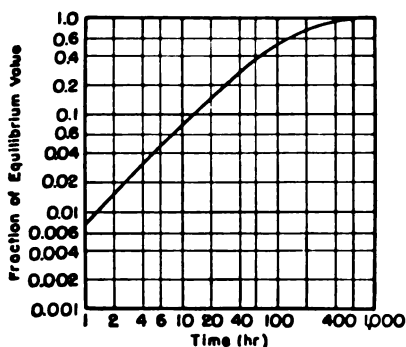


FIG. 16-2. Growth of radium daughters.

1. Add 100 ml concentrated nitric acid to the 24-hr urine sample in a suitable beaker and evaporate to dryness on a hot plate.
 2. Cool, and add about 20 ml additional nitric acid and again evaporate to dryness.
 3. Repeat the previous step until a white ash is obtained.
 4. Add 150 ml 0.1M nitric acid, warm for 5 min, and transfer the solution to a 250-ml centrifuge bottle.
 5. Centrifuge and transfer the supernatant to a second 250-ml centrifuge bottle.
 6. Rinse the beaker with 50 ml 0.1M nitric acid; use washings to wash the precipitate in the first centrifuge bottle.
 7. Centrifuge and combine both supernatants.
 8. To the combined supernatants add dropwise, while stirring mechanically with a platinum wire, 3 ml concentrated sulfuric acid followed by dropwise addition of 0.50 ml lead carrier solution (200 mg lead as nitrate per milliliter).
 9. Immerse the bottle in a bath of ice water and continue stirring for 15 min.
 10. Centrifuge and discard the supernatant liquid.
 11. Dissolve the precipitate by gently warming with 15 ml hydrochloric acid-ether reagent (consisting of 6 parts, by volume, of concentrated hydrochloric acid and 1 part diethyl ether).
 12. Transfer the solution quantitatively to a 50-ml centrifuge tube using a total of 10 ml hydrochloric acid-ether reagent for washing.
 13. Cool the tube in an ice bath.
 14. Add 0.40 ml barium carrier solution (10 mg barium as nitrate per milliliter) with stirring.
 15. Continue to stir for 5 min.
 16. Centrifuge in ice-packed cups for 5 min.
 17. Discard the supernatant solution.
 18. Add 1 ml of 0.1M nitric acid and expedite solution of the precipitate by stirring with a platinum rod.
 19. Transfer the solution with a capillary pipette to a platinum plate 4 cm in diameter ringed around the periphery with lacquer.
 20. Evaporate under gentle infrared heat.
 21. Add 4 drops 0.5M sulfuric acid and evaporate to dryness under infrared heat.
 22. Flame the plate gently to burn off the lacquer border; then flame to red heat.
 23. Alpha count.
- Coprecipitation with barium sulfate** is the basis for the following method which is reported to afford quantitative recovery of radium (Harley, J. H., and Foti, S., Determination of Radium in Urine, *Nucleonics*, vol. 10, no. 2, pp. 45-47, 1952):
1. Add aqueous ammonia to 200 ml urine in a 250-ml centrifuge bottle until precipitation is complete (pH about 7).
 2. Centrifuge and discard the supernatant liquid.
 3. Wash the precipitate with 25 ml 10 per cent aqueous ammonia.
 4. Centrifuge and discard the supernatant liquid.
 5. Dissolve the precipitate in a few milliliters acetic acid and add 200 ml distilled water.
 6. Bring the solution to a pH of 5 with aqueous ammonia using pH test paper.

7. Add 3 g solid ammonium sulfate and 1 ml 0.015M barium chloride (2 mg Barium); stir well, and allow to digest overnight.

8. Filter through fine filter paper and wash the precipitate with 0.5 per cent sulfuric acid.

9. Place the filter paper and precipitate in a small platinum dish and ignite at 800 to 900°C.

10. Add 2 ml 50 per cent sulfuric acid and 5 ml 48 per cent hydrofluoric acid; evaporate on a sand bath until the appearance of sulfur trioxide fumes.

11. Transfer to a 50-ml centrifuge tube with 40 ml distilled water and allow to digest overnight.

12. Centrifuge and discard the supernatant liquid.

13. Wash the precipitate with 10 ml distilled water; centrifuge and discard the supernatant liquid.

14. Wash the precipitate with 10 ml 95 per cent ethanol; centrifuge and discard the supernatant liquid.

15. Transfer the precipitate with ethanol to a nickel disk 2.8 cm in diameter; dry under infrared heat.

16. Alpha count.

URANIUM (FLUORIMETRIC)

Fluorimetric analysis for natural uranium is preferable to radiochemical or radio-metric analysis because of the low specific activity of natural uranium. The fluorimetric method (also known as the fluorophotometric or fluorometric method) offers high sensitivity (to about 5×10^{-10} g uranium per sample) and comparative freedom from interferences; it is thus the method of choice for work in radiation hygiene chemistry. No preliminary isolation or concentration of uranium is usually necessary; the sample (preferably in aqueous acid solution) is merely dried, fused with sodium fluoride (or suitable mixture containing sodium fluoride), and the fused mixture is exposed to ultraviolet excitation, the intensity of the emitted light being proportional to the amount of uranium present. If the presence is suspected of uranium isotopes possessing higher specific activity than normal, a radiochemical analysis is indicated.

Existing procedures for the fluorimetric analysis of uranium differ only in detail, and each comprises four steps: (1) sample preparation, (2) selection of flux, (3) fusion techniques, and (4) fluorescence measurement of the fused preparation.

Sample preparation involves bringing the sample into acid aqueous solution. Urine samples are acidified with either hydrochloric acid or sulfamic acid, preferably soon after collection; solid samples and blood may be ashed with mixtures of oxidizing acids to bring them completely into solution. Aliquots of the liquid sample (usually about 0.10 ml) are pipetted onto suitable platinum dishes; these dishes are lipped, and may be provided with a tiny handle to facilitate manipulation with tweezers or forceps. The sample is then dried in the dish by gentle heat, following which complete oxidation of the sample is assured by briefly flaming to red heat.

Flux (either as a preformed pellet or as a measured volume of solid) is then added to the sample in amounts varying from 0.10 to 2.5 g, depending upon dish design and the composition of the flux. For routine work on urines, the sample may be

pipetted directly onto the flux contained in the dish. The flux may be pure sodium fluoride, sodium fluoride containing 2 per cent lithium fluoride, or sodium fluoride containing variable percentages of sodium and potassium carbonates. High-carbonate fluxes melt at temperatures as low as 660°C; as a result, they may be fused in gold instead of the more expensive platinum, and their use is indicated where high-temperature fusion equipment is not available. Fluxes of pure sodium fluoride, or of sodium fluoride containing 2 per cent lithium fluoride, offer the greatest sensitivity and specificity; their use is recommended where temperatures of 1,000°C can conveniently be obtained.

Fusion of the flux and sample may be carried out in a muffle furnace for low-melting fluxes; for the carbonate-free fluxes, fusion by flame is expedient and advantageous, since the gas-burner atmosphere is an aid in the production of uniform fused mixtures. The dish is placed over a gas-air flame of sufficient intensity to melt the flux in 30 to 60 sec, the material is maintained in the molten state for 1 or 2 min, and is then allowed to cool undisturbed in a draft-free place. Precise reproduction of fusion conditions is probably the most important factor in securing good analytical results, since variations in flame composition, flame temperature, fusion time, and cooling time invariably give rise to erratic results. Each set of samples is accompanied by blanks and samples of known composition ("spikes") to assure the operator of the validity of his observed results.

The **fluorescence of the fused mixture** is then measured on an instrument designed to furnish the most efficient exciting radiation and to measure the emitted light. Several suitable instruments are described in the literature; a commercially available instrument is one manufactured by the Jarrell-Ash Company, Newtonville, Mass. With pure sodium fluoride fluxes, the fluorescence of the fused mixture must be measured in the fusion dish; with high-carbonate fluxes, it is often desirable to remove the button-shaped fused mixture and measure its fluorescence in a carrier designed for the purpose. Comparison of the fluorescence of the sample with that of standards affords a measure of the amount of uranium.

Quenching is a factor which may complicate interpretation of results; it is defined as any process occurring within a fluorescent substance that decreases the observed intensity of fluorescence. The presence of materials causing quenching has been observed in a small percentage of urines examined, and in routine work may be considered negligible. Samples which appear to contain significant amounts of uranium may be examined for the presence of quenchers by dilution or by spiking, both methods taking advantage of the fact that the extent of quenching is a function of the amount of quencher and is independent of the amount of uranium. The method of dilution is applicable to those samples whose uranium concentration is high enough to give conveniently measurable fluorescence with smaller aliquots; if fluorescent response is thus found to be proportional to sample size, obviously no quenching is occurring. The method of spiking involves adding a known excess (about tenfold) of uranium to the sample; the amount of added uranium being known, it is possible to measure the quenching effect of the sample upon the fluorescent response of the added uranium. This correction is applied to the observed fluorescence of the sample alone, yielding a figure of greater validity.

In the **analysis of blood for uranium**, it may be advisable to separate the iron (a quencher) before fluorimetric analysis; this is accomplished by ashing the blood,

chelating the iron, and purifying the uranium by ion-exchange technics. (Welford, G., and Alercio, J., Determination of Traces of Uranium in Blood, *USAE*C NYO-4000, 1952.)

PLUTONIUM

Analytical procedures for plutonium are carefully selected because of the absolute necessity of measuring with confidence very small amounts of the element. All the methods appearing in the literature for this determination involve one or more of the following steps: (1) carrying with a precipitate of bismuth phosphate; (2) carrying with a precipitate of lanthanum fluoride; (3) extraction with a solution of TTA (2-thenoyltrifluoroacetone) in an aromatic hydrocarbon; (4) extraction with chloroform in the presence of iron and cupferron (nitrosophenylhydroxylamine); and (5) electrodeposition, usually carried out as an ultimate step as the best means of preparing a very-low-mass film of rigidly defined area. The success of the first five steps above depends upon the plutonium's being present in one of its lower valences, presumably four; for this reason, chemical separations are usually carried out in the presence of hydroxylamine salts, sulfurous acid, or nitrite ion.

A comparatively **simple and rapid procedure for the determination of plutonium** in urine samples is given below; the chemical manipulations furnish a preparation of relatively low surface density suitable for alpha counting. The sensitivity attainable is obviously a function of counter background, counting time available, and the presence of trace alpha contamination in the reagents used; using counters averaging a background of about 0.05 counts/min, with 1-hr counting periods, the detection limit is about 0.5 disintegration/min/sample; the percentage recovery, or yield, averages about 80 per cent, and must be determined accurately by running suitable spikes (samples of known composition) so that an appropriate correction can be applied in reporting the results. The procedure isolates successfully actinium, thorium, neptunium, americium, and curium as well as plutonium (Schubert, J., Myers, L. S., Jr., and Jackson, J. A., Analytical Procedures of the Bioassay Group at ANL, *USAE*C ANL-4509, 1951):

1. Transfer a 24-hr urine sample (300 to 2,000 ml) to a 4-liter beaker. Rinse the original container with 100 ml of concentrated nitric acid, add the rinsings to the urine, cover with a watch glass, and evaporate to dryness.

2. Allow to cool. Add sufficient concentrated nitric acid to cover the solid matter. Evaporate slowly to dryness.

3. Repeat step 2 until the solid is perfectly white.

4. Drop 2 ml of concentrated nitric acid over the solid matter and let it soak in. Add 35 ml of water to dissolve the solid. A small amount of insoluble matter remains suspended in the solution.

5. Transfer the solution to a 50-ml conical-end centrifuge tube and centrifuge at 2,000 rpm for 5 min.

6. Decant the clear supernatant solution into a 90-ml round-end pyrex centrifuge tube.

7. Wash down the sides of the 50-ml tube with 1 ml of concentrated nitric acid and then with 2 ml of water. Slurry the insoluble matter at the bottom of the tube to leach out any remaining alpha activity. Centrifuge. Pour the supernatant liquid into the main solution contained in the 90-ml centrifuge tube.

8. Dilute the solution to 80 ml, add 2 ml saturated sulfurous acid, stir, and allow to stand 20 min at room temperature.

9. Place in a water bath at 85 to 90°C and stir vigorously with the aid of a stirring motor. Allow the solution to reach the bath temperature. Add dropwise 1 ml of the bismuth carrier solution; this contains 100 mg Bi ion/ml of 10*N* nitric acid.

10. Add 1 ml of concentrated phosphoric acid dropwise and slowly.

11. Let the precipitate of bismuth phosphate digest with constant stirring at 85 to 90°C for 1 hr.

12. Stop the stirring, remove the centrifuge tubes from the water bath, and let the precipitate settle. The tube may be centrifuged if necessary.

13. Suck off the supernatant solution through a slender glass tube connected to a water aspirator.

14. Transfer the precipitate and the 1 or 2 ml of remaining liquid to a 15-ml conical-end centrifuge tube. Wash down the 90-ml tube with distilled water, and add this to the 15-ml tube.

15. Centrifuge the 15-ml tube and discard the supernatant solution. Wash the 90-ml tube with 10 drops of concentrated hydrochloric acid followed by 10 drops of water. Add this liquid to the precipitate in the 15-ml tube. Repeat, using 10 drops of hydrochloric acid followed by 10 drops of water. Stir until the precipitate dissolves.

16. Dilute the solution to 5 ml, add 0.5 ml of lanthanum carrier solution (0.5 mg lanthanum/ml 1*N* nitric acid), and stir with a very thin platinum rod. Add 0.5 ml of 25 per cent HF and stir well.

17. Immediately centrifuge for 5 min, discard the supernatant solution, wash the precipitate with 6 ml of water, centrifuge, and discard the supernatant. Transfer the precipitate to a stainless-steel counting disk with 2*N* nitric acid.

18. Evaporate the sample slowly to dryness without spattering. Flame the disk briefly to remove the last traces of organic matter.

19. Count.

The following procedure for the **determination of plutonium** in urine furnishes the most satisfactory results; it is not economical of time or effort, but painstaking chemical separation combined with the nuclear-track technique affords a 95 per cent recovery of plutonium, and the detection limit is 0.05 disintegration/min at a very high confidence level (Schwendiman, L. C., Healy, J. W., and Reid, D. L., Nuclear Track Emulsions Applied to the Determination of Plutonium, *USAEC HW-22680*, 1951):

1. To the urine sample contained in a 1,500-ml Erlenmeyer flask add concentrated nitric acid until a definite color change is observed; the final acid normality will be about 2 to 4. Add 5 ml of octyl alcohol to minimize foaming and evaporate to dryness.

2. Transfer the dried salts to a 250-ml beaker using 8*N* nitric acid followed by sufficient water, and carefully evaporate to dryness.

3. Muffle at 600°C for 1 hr.

4. Add 50 ml 2*N* nitric acid and allow to stand overnight or until salts dissolve.

5. Add 0.5 g hydroxylamine hydrochloride and stir until dissolved.

6. Transfer to a 100-ml pyrex centrifuge tube. Wash the beaker with small volumes of water, adding washes to the centrifuge tube.

7. Add 1 ml lanthanum solution (20 mg lanthanum ion/ml 2N nitric acid) and stir.
8. Make volume to 75 ml with 2N nitric acid, and stir thoroughly.
9. Add 5 ml 27N hydrofluoric acid.
10. Stir and let the sample stand for 2 min.
11. Centrifuge at 2,500 rpm for 2 min.
12. Discard the supernatant, draining the tube carefully to remove as much of the liquid as possible.
13. Slurry the precipitate in 1 or 2 ml concentrated nitric acid. Add 6 to 10 ml water, then add 2N nitric acid in 10-ml portions and with constant stirring until the total volume is 75 ml.
14. Add 5 ml 27N hydrofluoric acid.
15. Repeat steps 10, 11, and 12.
16. Dissolve the precipitate in 40 ml aluminum nitrate solution (74 per cent aluminum nitrate nonahydrate in 0.45N nitric acid) by first slurrying with 1 to 2 ml of solution to break up lumps and then by addition of 10-ml portions with repeated agitation.
17. Pour the resulting solution into a 125-ml separatory funnel.
18. Wash the centrifuge tube with exactly 10 ml distilled water, pour into the separatory funnel and mix thoroughly.
19. Add 0.25 ml 2M sodium nitrite solution and allow to stand for 15 min.
20. Add 10 ml TTA solution (5 per cent 2-thenoyltrifluoroacetone in benzene) to the separatory funnel.
21. Shake the funnel for 20 min and allow to stand until the phases separate. Release the pressure after the first few minutes of shaking.
22. Discard the aqueous phase and wash with 20 ml distilled water by shaking for 10 min. Discard the wash.
23. Repeat using 10 ml distilled water and discard the wash.
24. Add 10 ml 8N hydrochloric acid and shake for 20 min.
25. Allow the phases to separate; collect the acid layer in a 30-ml beaker.
26. Wash the benzene layer with 5 ml 8N hydrochloric acid, shaking for 5 min. Add this acid to the previous 10 ml of acid; discard the benzene layer.
27. Evaporate the 8N hydrochloric acid without boiling until only 1 or 2 ml remain.
28. Neutralize the remaining acid with 12N potassium hydroxide solution.
29. Add 2 ml sodium hypochlorite solution (5 to 6 per cent available chlorine), 5 ml 2N potassium hydroxide, and evaporate carefully until the volume is reduced to one-half the original.
30. Transfer to an electrodeposition cell with distilled water to a final volume of 10 ml.

The plastic electrodeposition cell has an internal diameter of $\frac{5}{8}$ in. and is $2\frac{5}{8}$ in. long. It is threaded at the bottom to accept a stainless-steel cap which holds a $\frac{1}{2}$ -in. stainless-steel disk in contact with the constricted bottom of the plastic cell wall so that deposition occurs on a 7-mm-diameter area in the center of the disk. The cap also serves as a cathode connection.
31. Plutonium is electrodeposited on the disk by applying for 5 hr a current of 80 ma at 12 volts between the cathode cap and a platinum rod extending down into the solution.

32. The platinum anode is removed from the cell with the current on. The solution is discarded and the cell is rinsed with distilled water. The stainless-steel disk bearing the electrodeposit is removed.

33. Instead of counting on a proportional counter or scintillation counter, the disk is evaluated for radioactive material present by exposing it to a nuclear-track emulsion for a week, developing the emulsion by standard techniques, and examining fractions of the exposed area of the film with a microscope for the presence of tracks made by alpha particles emitted from the disk. The procedure is lengthy and detailed; considerable special equipment is necessary to ensure accurate registry of the disks on the nuclear-track emulsion and to facilitate measuring the alpha tracks over a specified fraction of the exposed surface. Further details may be ascertained by reference to the original publication (Schwendiman, L. C., Healy, J. W., and Reid, D. L., Nuclear Track Emulsions Applied to the Determination of Plutonium, *USAEC HW-22680*, 1951).

THORIUM

Natural thorium in ashed biological samples may be isolated by coprecipitation with lanthanum fluoride or calcium oxalate, or by extraction with TTA in benzene. Many other isolation techniques are available but are not common to procedures in radiation hygiene chemistry.

It is not convenient to measure radiometrically the isolated natural thorium, since the specific activity is low and since there are complicating factors in the decay scheme; the final measurement is therefore made by colorimetric or spectrophotometric means. Reagents used for this purpose are morin (2',3,4',5,7-pentahydroxyflavone) and thorin [disodium salt of 2-(2-hydroxy-3,6-disulfo-1-naphthylazo) benzenearsonic acid]. Morin offers the possibility of making more sensitive measurements, since the thorium-morin complex possesses high absorptivity; however, morin is usually available only in notably impure form and requires careful purification before use.

The following procedure, capable of detecting 1 μg of thorium per sample, calls for the **measurement of thorium with morin** after isolation (Perkins, R. W., and Kalkwarf, D. R., Determination of Thorium in Urine, *Anal. Chem.*, vol. 28, pp. 1989-1993, 1956):

1. To a 500-ml urine sample contained in a 1,000-ml Erlenmeyer flask add 2 ml lanthanum solution (10 mg lanthanum/ml) and 100 ml concentrated nitric acid and evaporate to a volume of 25 to 25 ml.
2. Allow the sample to cool for 5 min.
3. Transfer the sample to a 100-ml Lusteroid centrifuge tube containing 10 to 20 ml distilled water. Wash the flask with three 10 to 15-ml portions 2*N* nitric acid, adding washings to the centrifuge tube.
4. Add 5 ml concentrated hydrofluoric acid, stir thoroughly, and allow to stand for 1 hr.
5. Centrifuge at 1,600 rpm for 2 min. Decant the solution, taking care to leave the precipitate undisturbed, and dissolve the precipitate in 10 to 20 ml 2*N* nitric acid directed in a fine stream from a wash bottle; dilute to 80 to 90 ml with 2*N* nitric acid.

6. Add 5 ml concentrated hydrofluoric acid, stir thoroughly, and allow to stand for 5 min.

7. Centrifuge at 1,600 rpm for 2 min.

8. Decant the solution and transfer the precipitate to a 45-ml Vycor evaporating dish with the aid of 10 to 15 ml 2 *N* nitric acid.

9. Add 3 ml concentrated perchloric acid and evaporate to dryness on a hot plate.

10. Dissolve the residue in 3 ml *N* nitric acid and transfer the solution to a separatory funnel containing 10 ml TTA solution in benzene (10 per cent). Wash the Vycor dish with three 1-ml portions distilled water, adding the washes to the separatory funnel.

11. Shake for 5 min at maximum agitation on a mechanical shaker.

12. Add 10 ml distilled water and shake for 15 min on the mechanical shaker.

13. Discard the aqueous layer.

14. Wash the benzene layer by shaking it with three 5-ml portions of 0.2*N* nitric acid for 5 min each, discarding the aqueous layer each time.

15. Wash the inside of the separatory funnel with 25 to 35 ml distilled water by spraying from a wash bottle and discard the wash water.

16. Add 10 ml 2*N* nitric acid and shake mechanically for 15 min.

17. Let stand for 30 min to allow complete separation of the phases; incomplete separation here may lead to a mild explosion and loss of the sample in succeeding steps.

19. Withdraw the aqueous phase into a 45-ml Vycor dish containing 3 ml concentrated perchloric acid. Evaporate the samples to dryness, using care to prevent the entrance of dust from the hood ventilating air.

19. Allow the dishes to cool.

20. Add 3.3 ml dilute perchloric acid (perchloric acid diluted with distilled water to pH 2.0), swirl for ½ min, allow to stand 5 min, and swirl again.

21. Transfer 3.0 ml of the sample to the spectrophotometer cell and add 0.50 ml morin solution (0.025 per cent in absolute ethanol).

22. Prepare a blank solution containing 3.0 ml dilute perchloric acid (described in step 20) and 0.50 ml morin solution. Place the cap on the cells and invert several times to mix.

23. Allow to stand for 1 hr and measure the absorbance at a wavelength of 412 millimicrons, using a slit width of 0.03 mm on a Beckman Model DU spectrophotometer.

The following procedure involves the **measurement of thorium with thorin** following partial isolation from the bulk of the accompanying urine solids (Albert, R., Klevin, P., Fresco, J., Harley, J., Harris, W., and Eisenbud, M., *Industrial Hygiene and Medical Survey of a Thorium Refinery*, *A.M.A. Arch. Ind. Health*, vol. 11, pp. 234-242, 1955):

1. To a 200-ml urine sample, add concentrated aqueous ammonia to pH 8.

2. Centrifuge, decant the supernatant liquid, and discard.

3. Add 10 ml concentrated hydrochloric acid to the precipitate, followed by 100 ml water.

4. Transfer the solution to a 400-ml beaker and wet ash with 100 ml concentrated nitric acid and 1 ml perchloric acid.

5. Add 10 drops concentrated hydrochloric acid to the white residue, followed by water to a final volume of 100 ml.
6. Adjust the pH of the solution to 2.5 with dilute ammonia or dilute hydrochloric acid, using a pH meter.
7. Heat the solution to boiling.
8. Add drop by drop 4 per cent ammonium oxalate solution. Approximately 7 ml is required to precipitate the calcium usually present in the 200-ml urine sample.
9. To the hot solution, add very slowly from a burette a filtered dilute solution of ammonia (6 per cent) at about 5 ml/min until the solution is neutral or slightly alkaline.
10. Let the solution stand for 1 hr in a warm place to ensure complete precipitation of calcium oxalate.
11. After the precipitate has settled, test the solution with a few drops of ammonium oxalate solution for complete precipitation.
12. Decant the clear supernatant liquid through a filtering crucible (Selas 3010).
13. Transfer the precipitate quantitatively to the crucible using a wash bottle and rubber policeman.
14. Wash the precipitate with cold, very dilute (0.25 per cent) ammonium oxalate solution ten times to ensure removal of chloride ion.
15. Dry the precipitate at 110°C for 1 hr; then place the crucible in an electric muffle at 500°C for 2 hr.
16. Cool the crucible in a desiccator and weigh. The precipitate should now be calcium carbonate.
17. Place the crucible in a Gooch filtering funnel with the outlet leading into a 15-ml graduated centrifuge tube.
18. Add from a pipette 1 ml concentrated hydrochloric acid, allowing the acid to run down the sides of the crucible.
19. Turn on gentle suction, and wash the crucible with a very fine stream of water until the volume of solution in the centrifuge tube is approximately 6 ml.
20. Dry and weigh the empty crucible.
21. Remove the test tubes from the filtering funnels, add 2 ml 10 per cent hydroxylamine hydrochloride solution, and heat gently to boiling.
22. Add 12 drops concentrated hydrochloric acid (or until pH is approximately 0.8).
23. Add 1 ml 0.1 per cent aqueous thorin solution.
24. Dilute to 10 ml.
25. Measure the absorbency at a wavelength of 545 millimicrons on a Beckman DU spectrophotometer, using 5-cm cells.
26. Reference solutions for preparation of calibration curves must contain, on the average, the same amount of calcium as the samples.

Thorium-230 (Ionium) may be determined in urine or in other materials by any procedure which isolates the thorium in a pure enough state for alpha counting. In the first procedure under thorium, the aqueous phase after step 17 may be evaporated and transferred to a suitable disk or planchet; alternatively, the first procedure under plutonium may be used to isolate thorium in a state suitable for alpha counting.

Dibutylphosphoric acid (DBP) is a powerful extractant for thorium, but its use is limited because of the difficulty of reextraction from the solvent and the impossibility of simply evaporating the DBP. However, a procedure for the determination of thorium in bone has been described whereby small enough quantities of DBP are used so that the extract can be ashed directly on a disk or planchet. Excellent recoveries and decontamination are afforded.

POLONIUM

Polonium in solution is characterized by its tendency to undergo spontaneous electrochemical exchange with various metal surfaces, notably silver, copper, and nickel (in practice, alloys of these metals may be used); the ability of polonium to form a dithizonate has not been exploited for analytical use because electrochemical exchange affords at once a clean easy separation and a highly desirable counting preparation.

Methods for determining polonium are the same in principle and differ only in detail. Exposure of the sample to the metal surface is made at elevated temperatures with stirring; for routine work, baths are operated well below the boiling point to minimize evaporative loss. The speed of stirring is not critical. Nitric acid interferes with the reaction and must be removed before the plating procedure. Ferric iron in concentrations above $M/1,000$ interferes and is removed by the addition of ascorbic acid; this addition is necessary in the analysis of ashed blood samples but is not indicated in routine work on urines. It is expeditious to stir the metal disk in the sample solution, but greater sensitivity may be realized by carrying out the exchange in a suitable plating cell whereby the polonium activity appears on but one side of the disk.

The **sample** must be suitably pretreated so that the polonium present is in solution in dilute hydrochloric acid (usually 0.5*N*) before electrochemical exchange is allowed to occur. Solid samples (tissue, feces, filter paper, etc.), as well as blood, must first be ashed, preferably by wet digestion with nitric acid or nitric-perchloric acid mixtures. Oils and organic solvents may be extracted with dilute hydrochloric acid, by which means polonium is quantitatively transferred to the aqueous layer. Urines are not usually ashed or digested, especially for routine work; it is merely necessary to be certain that polonium is not lost from solution between the times of sample collection and analysis; acidification of the sample suffices.

The **routine examination** of urines suspected of containing not less than about 5 disintegrations/min of polonium per liter is carried out as follows (Milligan, M. F., Campbell, E. E., Eutsler, B. C., McClelland, J., and Moss, W. D., *Analytical Procedures of the Industrial Hygiene Group, USAEC LA-1858*, 2d ed., 1958):

1. The urine sample should exceed 100 ml in volume; should more than 1 hr elapse before analysis, sulfamic acid (1 mg/ml) is added and the sample is refrigerated.
2. Measure 100 ml of the sample with a graduate and pour into a 150-ml beaker; rinse the graduate with 20 ml of 6*N* HCl and add to the urine. Place the beaker containing the acidified sample in a constant-temperature bath at 50 to 55°C.

3. Degrease a nickel disk in trichloroethylene; dip in concentrated nitric acid, rinse in distilled water, dip in concentrated hydrochloric acid, and rinse in distilled water. Be sure that the surfaces are bright. The nickel disks are made from 0.025-in. nickel sheet; they are $\frac{3}{8}$ in. in diameter, with a $\frac{1}{8}$ -in. hole set $\frac{1}{16}$ in. from the edge.

4. Suspend the disk on a glass stirring hook in the urine and stir mechanically for 1 hr. Speed of agitation is not critical.

5. Remove the glass stirring hook and the disk from the solution; rinse down both the glass rod and the disk with a fine stream of distilled water; hang the assembly to air-dry the disk.

6. Count both sides of the disk to obtain the total polonium activity.

A more sensitive technique for the examination of larger volumes of urines which may contain as little as 0.05 disintegration/min of polonium per liter is as follows (Black, S. C., Low Level Polonium Determination in Tissue and Urine, *USAE C* UR-463, 1956):

1. Acidify the urine by adding 50 ml of concentrated hydrochloric acid per liter.

2. Add nickel powder in the amount of 2 g/liter.

3. Add 100 mg of ascorbic acid and stir rapidly for 2 hr at a temperature of 90 to 100°C.

4. Filter through a medium-speed filter paper with suction.

5. Place the filter paper back in the original container and pour aqua regia, 30 ml/2 g of nickel powder, through the funnel over the paper and powder and wash once with 25 ml of 0.5*N* HCl.

6. Boil the solution 2 to 3 hr to dissolve the paper and the powder; then add 20 ml of a 2:1 mixture (v/v) of 70 per cent perchloric acid and concentrated nitric acid.

7. Boil down until dense white fumes appear; add cautiously 1 ml of concentrated hydrochloric acid; if no brown fumes are evolved, all the nitric acid has been destroyed.

8. Following complete destruction of the nitric acid, make the solution alkaline to phenolphthalein with saturated NaOH solution; then make the solution 0.5*N* in acid, using concentrated HCl.

9. Cover the container and reflux gently for 2 hr to bring into solution any polonium which may have deposited on the glass during the alkaline stage.

10. Transfer the solution to a plating cell fitted with a cleaned silver-foil disk, using three rinses of 0.5*N* HCl so that the total volume does not exceed 200 ml. The plating cell is a Davol-brand nursing bottle with the bottom cut off. A $\frac{1}{16}$ - by 1 $\frac{1}{2}$ -in. neoprene disk and a 0.005- by 1 $\frac{1}{2}$ -in. silver-foil disk fit inside the cap, which is then firmly screwed on the bottle. The silver foil is cleaned by scrubbing with a mild household abrasive powder until the metal surface is clean and bright.

11. Add 100 mg of ascorbic acid.

12. Place in a water bath at 90 to 100°C and stir rapidly for 2 hr.

13. The supernatant solution is decanted and the cell is rinsed gently with distilled water; the foil is removed, blotted dry with absorbent tissue, and counted for total polonium activity.

The analysis of other materials than urine for polonium activity is carried out by one of the two methods indicated, following suitable ashing, extraction, or digestion procedures.

RUTHENIUM

A method for the **determination of Ruthenium-106** has been developed because of the apparent tendency of this fission product to concentrate in plant tissues (the method can be applied with suitable modifications to water, soil, or ash samples). Ruthenium is reduced in acid solution from valences of 3, 4, 6, or 8 to the elementary state, a procedure affording 91 per cent recovery (Kennedy, M. R., and Fitzgerald, J. J., Analysis of Ruthenium-106 in Vegetation, *USAEC KAPL-1052*, 1954):

1. Muffle 5 g of vegetation in a 100-ml beaker overnight at 500°C, or until the ash is white.
2. Add 20 mg ruthenium carrier and 20 ml 6*M* hydrochloric acid to the ash.
3. Heat to boiling and centrifuge.
4. Transfer the supernatant solution to a 100-ml beaker; rinse the precipitate and centrifuge; add the washings to the beaker.
5. Evaporate the solution to approximately 10 ml and transfer to a 100-ml centrifuge tube.
6. Stir vigorously with an electric stirrer and heat to boiling with a gas flame.
7. Stir continuously while adding magnesium powder slowly until all the ruthenium is reduced; at this point, foaming ceases and silvery particles of unreacted magnesium may be seen in the solution.
8. Slowly add 6*M* hydrochloric acid until foaming stops; continue stirring the solution and add sufficient 6*M* hydrochloric acid to bring the volume to 50 ml.
9. Boil the mixture 2 min with continuous stirring.
10. Filter the precipitate through a Hirsch funnel fitted with Whatman 40 filter paper. Dry the precipitate with alcohol, ether, and air.
11. Counting is done through absorbers to correct for the weak betas from Ruthenium-103 and from any zirconium and niobium which may have been carried through the procedure.

Section 17

EQUIPMENT FOR HANDLING, STORAGE, AND TRANSPORTATION OF RADIOACTIVE MATERIALS

By

MYRON B HAWKINS, *Institute of Engineering Research, University of California,
Richmond Field Station, Richmond 4, California*

Basic Principles.....	17-2
Radiation-exposure Control.....	17-2
Contamination Control.....	17-3
Properties of Radioactive Materials.....	17-5
Handling Techniques.....	17-6
Direct-viewing Equipment.....	17-7
Tongs and Manipulators.....	17-7
Remote-control Unit-process Equipment.....	17-10
Close Shields.....	17-11
Contamination-control Equipment.....	17-13
Shielding-barrier Equipment.....	17-15
Closed-cell Equipment.....	17-23
Underwater Facilities.....	17-31
Storage Equipment.....	17-32
Transport and Shipment.....	17-32
Special Shipping Equipment.....	17-34

EQUIPMENT FOR HANDLING, STORAGE, AND TRANSPORTATION OF RADIOACTIVE MATERIALS

Myron B Hawkins

BASIC PRINCIPLES

See page 17-35 for General References.

The handling of radioactive materials requires the use of special techniques, equipment, and facilities to protect personnel from the biological effects of the radiations. Such precautions may be required, in addition, to protect radioactive materials from cross contamination and radiation-detection devices from extraneous radiation fields. The presence of radioactive material does not invariably result in a radiologically hazardous or undesirable condition. The amount of radioactive material present, the type and energy of the radiation, and the physical and chemical state of the radioactive material (and its diluent) all influence the degree of the potential hazard, and consequently the handling problem. The problem has two parameters: (1) the control of external radiation dosage, and/or (2) the control of contamination.

RADIATION-EXPOSURE CONTROL

Maximum permissible exposures to external radiation are expressed in dose per unit time. Most standards employ the week as the basic time unit, e.g., 0.3 r/week or 0.3 rad/week. It is sometimes considered permissible to accumulate in shorter periods the dose normally allowed for a period up to 13 weeks. This technique allows a more economic and practical approach to the utilization of the permissible exposure of personnel. In any case, the control of radiation exposure should be based on a sensible and concurrent utilization of the three basic techniques of dosage control: distance, time, and shielding.

Distance control reduces the dose rate by increasing the distance from the recipient to the radiation source. Not only is the flux reduced because of geometric considerations, but in many cases (e.g., with alpha and beta radiations) the absorption of the radiation by air can be significant. Very seldom is it permissible to grasp radioactive materials (except ores and materials of similar low activities and concentrations) directly with the hands. Thin-wall containers seldom provide enough shielding to permit direct handling. Distance control is accomplished by the use of forceps, tongs, or other remote-control devices.

Shielding reduces the radiation dose rate by interposing absorbing materials between the radioactive source and the worker. The thickness of shielding required

is a function of the type of radiation present, its energy and quantity. Techniques for calculating the effectiveness of various shielding materials are found in Sec. 8. The design of all shields should consider the requirement for attenuation or elimination of the secondary as well as the primary radiations. General information on the practical aspects of shielding is included throughout this section.

Shields are designed either to fit closely around the source (i.e., close shields) or closely around the worker (e.g., gloves, lead aprons) or to be a fixed barrier in the space between the two. The close shield and personnel shields tend to immobilize the source and/or the worker. Barriers usually require indirect-viewing and remote-manipulation techniques.

Exposure-time control reduces the radiation dosage by limiting the time in the radiation field. Thus, work of short duration can be done in high-radiation fields without overexposure to personnel if their exposure time is controlled. Personnel are often "rotated" if the work is of long duration. Obviously, the use of exposure-time control lessens the requirement for shielding and remote-control or manipulative equipment. In fact, the use of such equipment on an intricate job can often result in greater exposures than the more direct approach of using time control with direct viewing and manipulation.

Operations in radiation fields greater than that required to give permissible dosages with full-time occupancy should always be planned and rehearsed to eliminate nonproductive periods and therefore unnecessary exposure.

The technique of utilizing exposure-time control to limit dosage is best applied to operations that are:

1. Nonroutine
2. Of relatively short duration and/or utilizing short-half-life materials
3. In radiation fields low enough to be practical, i.e., with a permissible exposure period of at least 30 sec, preferably much greater

CONTAMINATION CONTROL

Contamination control becomes a problem when radioactive materials are in physical and chemical forms that allow them to become dispersed from the main body of material. Such dispersal may be caused either by accidents (e.g., explosions, spills, leakage) or by failure to apply adequate precautions to prevent inadvertent loss during handling and processing. As established by regulation, permissible levels of contamination are expressed for surfaces in number of disintegrations/minute/unit area for alpha emitters and number of rads/hour for beta and gamma emitters. Permissible concentrations in air, water, and solids are expressed in microcuries/unit volume. The actual numerical values vary with the various radioisotopes and in some cases with their chemical form. Aside from health-hazard considerations, the possibility of cross contamination of experiments may be of prime importance in establishing permissible levels and the type of contamination control utilized.

The two basic techniques of contamination control widely used are "dilution and dispersion" and "containment and concentration."

The **dilute, disperse, and decontaminate** approach to contamination control places the radioactive material and the worker in the same physical environment.

17-4 EQUIPMENT FOR HANDLING, STORAGE, AND TRANSPORTATION

High ventilation rates and special clothing protect the worker. Operations likely to produce air-borne contamination are performed in highly ventilated areas. Operations during which spillage or breakage might occur are performed over surfaces covered with absorbent material. Frequent radiological monitoring is required to detect inadvertent dispersion. Since contamination of the workers' environment is anticipated, surfaces should be compatible with decontamination procedures (Sec. 19). Fume-hood systems are designed to filter and dilute the air-borne contaminant to levels compatible with waste-disposal regulations. Because of the probable dispersal of material, routine monitoring of ducts, drainpipes, and many unsuspected locations is required.

The dilute, disperse, and decontaminate philosophy is satisfactory and sensible for many applications. While by no means limiting, the following criteria, individually or collectively, may facilitate the use of some variation of this approach:

1. Relatively small quantities of radioactive material
2. Short half-life
3. Low biological hazard
4. Low contamination potential in the process or the state of the material
5. Only sporadic use of radioactive material

With some notable exceptions, the dilute, disperse, and decontaminate approach to contamination control allows many operations to be done easily and rapidly and with a minimum of expenditure for special equipment. However, if "dilute and disperse" is employed at "high" levels, laboratory construction can become quite expensive.

The **contain and concentrate** philosophy of contamination control basically places one or more physical barriers between the worker and the radioactive material so that they are in separate atmospheric environments. Such a barrier encloses the material itself quite closely (such as a capsule), or encloses the material during a process (as in vacuum-line equipment) or encloses the material and the process equipment (as in gloved boxes or manipulator cells). Such barriers are completely airtight or nearly so. The latter type of area is maintained at lower pressures than the inhabited areas in order to prevent the loss of any air-borne contaminant. The air used to ventilate such spaces is usually filtered or otherwise processed to avoid discharging excessive quantities of contaminant.

The purpose of the barrier is, of course, to confine any loose contaminant to an environment or area that is not normally inhabited. Thus, even in the case of accidents and spills, both personnel and experiments are protected. It should be noted that the rehabilitation of such an area usually requires the employment to some extent of the dilute, disperse, and decontaminate philosophy.

The employment of the "containment" approach often limits the flexibility, ease, and convenience of operations since remote control is usually implied. Equipment, particularly when radiation-dosage control is also involved, can be complicated and expensive. Criteria that individually or collectively may require or influence the use of the "containment" approach to contamination control are:

1. Large quantities of radioactive materials
2. Materials of high toxicity
3. Long half-life
4. Repetitious use of radioactive materials

5. Repetitious use of the processes involved

6. High contamination potential of the processes and/or radioactive materials

While in many cases the "containment" approach makes operations more difficult and results in added expense, the opposite is also often true since the use of such equipment can be substituted for expensive building modifications.

PROPERTIES OF RADIOACTIVE MATERIALS

The relative hazard of the radioactive materials determines not only the degree but the type of protection and handling equipment required. Since the type of radiation emitted is a dominating characteristic in many instances, the following paragraphs describe the influence of the radiation on equipment design. In addition, various specific radioactive materials that present special and/or complex handling problems are described.

Beta particles, i.e., electrons, are emitted by various radioisotopes in a continuous energy spectrum from zero to a maximum energy that is characteristic of the particular isotope. These particles are relatively easily absorbed, having a maximum range in air of a few meters (~ 8 m for beta particles of 2-Mev maximum energy) or a few millimeters in a more dense material (~ 1 cm of water for a 2-Mev maximum energy). Absorption of beta particles results in part in the production of bremsstrahlung, i.e., X-rays of a continuous spectrum. The fraction of the beta energy converted to bremsstrahlung is proportional to the number and energy of the beta particles and the atomic number of the absorber. Thus, absorbers of a low atomic number should be used for beta absorption, particularly for high-energy beta particles and when large quantities of radioactive material are involved.

A common practice for the shielding of beta radiation involves the use of plastic or aluminum of sufficient thickness to absorb the primary radiations completely. Additional gamma shielding (for the bremsstrahlung) is added if required. Very-low-energy betas such as those from C-14, S-35, and H-3 are completely absorbed by rubber gloves.

Because beta particles are absorbed by relatively thin layers of absorber, beta-emitting materials are susceptible to self-shielding by diluents. For instance, liquids may emit only a fraction of the radiation flux that is emitted by the same quantity of radioactive material in evaporated state. The exploitation of self-absorption by dilution is an important handling technique. However, it should be noted that the reverse process (e.g., evaporation) can increase the radiation dose rate manyfold.

Positrons are emitted by some radioisotopes. The design of shielding for the primary radiation is similar to that for beta particles except that the annihilation radiation (i.e., two 0.51-Mev gamma photons for each positron) must be considered.

Gamma radiation is emitted during the disintegration of many radioisotopes. Gamma photons are basically penetrating, the most efficient absorbers for gamma photons being heavy elements such as lead and uranium. However, steel, ordinary concrete, and "heavy" aggregate concrete are commonly used for large installations. Air attenuation of gamma radiation being negligible, distance factors are computed on the basis of the inverse-square law. For relatively large quantities of gamma-emitting materials, the scattering of radiation during absorption or off walls and

17-6 EQUIPMENT FOR HANDLING, STORAGE, AND TRANSPORTATION

ceilings becomes a problem that must be considered in the design of the handling equipment.

Alpha particles are easily absorbed by very thin absorbers (such as the outermost layer of skin). Their range in air is a few centimeters; consequently, natural alpha radiation provides no external radiation problem. (Note: Accelerated alpha particles can penetrate the dead layer of skin and cause damage.) However, alpha-emitting materials can be extremely hazardous if inhaled, injected, or otherwise taken into the body. Thus, the basic handling problems for alpha-emitting materials are those concerned with contamination control.

Neutrons are produced by nuclear reactors, nuclear generators, and specially prepared sources. The latter utilize mixtures of beryllium and radium or polonium to produce an alpha-Be reaction that results in the production of neutrons of an average energy of about 4 Mev. Other sources use the gamma photons from radium or other gamma emitters to produce photoneutrons from beryllium or deuterium oxide. The latter sources may emit significant gamma-radiation fields.

Air attenuation of neutrons is negligible.

Shielding of neutrons can be accomplished by the use of materials of a high hydrogen (or other light element) content (e.g., paraffin, water, or concrete) to slow the neutrons down, with another material (e.g., boron or cadmium) to absorb the resulting thermal neutrons, and a final material to absorb the resulting radiation (e.g., lead for the gamma photons from cadmium). However, lead alone is often used.

Radium and its decay products emit significant quantities of alpha, beta, and gamma radiations. It is considered one of the more hazardous radioactive materials since (1) the inhalation-ingestion hazards are high, (2) a radioactive gas, radon, is released as the initial decay product, (3) the beta radiations are significant from a contact-dose standpoint, (4) the gamma radiations can create external radiation problems, (5) the half-life is long, and (6) the specific activity is usually high.

Radium radiation sources are permanently sealed to prevent escape of the radon gas. The wall thickness of therapeutic sources is usually a minimum of 0.5 mm of platinum to absorb the beta radiations completely.

Handling techniques for unsealed radium and its daughters must consider contamination control. The radium and radon series because of their beta and gamma emissions require consideration of radiation-dose control.

HANDLING TECHNIQUES

As the degree of radiation and/or contamination control increases, equipment generally becomes more complex and more expensive. Although many factors influence the degree of complexity, and in no case is the distinction clear-cut, the following categories tend to separate different techniques (and equipment) for handling radioactive materials:

1. **Direct-viewing** techniques for low radiation levels
2. **Shielding-barrier** techniques for intermediate levels
3. **Closed-cell** techniques for high radiation levels
4. **Underwater** techniques

DIRECT-VIEWING EQUIPMENT

When radiation dose rates are low enough, the most efficient operation method is that of **direct viewing**. Shielding barriers are not employed, and the worker has a direct unobstructed view of the source and/or its container.

The technique utilizes:

1. **Distance control** as provided by:
 - a. Straight tongs and manipulators
 - b. Remote-control devices such as unit-process manipulators (e.g., pipettes)
 - c. Unit-process glassware (e.g., vacuum-line apparatus)
2. **Shielding**
 - a. That provided by the source container and handling equipment (including gloves)
 - b. Close-fitting radiation shields for containers
 - c. That inherent in the bulk of the process material
3. **Exposure-time control**

Direct-viewing operations can be performed in the open, e.g., outdoors if the contamination potential is low. Otherwise, they should be performed in an adequately ventilated area (e.g., fume hood) or in some cases in enclosed hoods or gloved boxes. In the latter, the use of contamination barriers does not constitute a violation of the direct-viewing criteria.

Occasionally, particularly in emergency conditions, **contamination control** may consist of the use of special clothing and masks for the protection of operating personnel. Lead-impregnated gloves, aprons, etc., are used in connection with X-ray equipment (or other operations involving low-energy photons) to provide shielding. Such protective clothing is described in Sec. 18.

TONGS AND MANIPULATORS

The basic tools for **distance control** are forceps, tongs, and manipulators. These devices provide the operator with an approximate duplication of the action of his own hands and fingers in grasping, transporting, and releasing objects that have a wide range of sizes, shapes, and weights. As distances become greater or the objects become heavier or the movements become more complex or the precision of positioning becomes greater, manipulations become more difficult and more hazardous operationally.

Standard laboratory tongs and forceps designed for general use are often quite satisfactory for work with radioactive materials at short ranges. Spring-type forceps (Fig. 17-1a) of various types are available in lengths from 4 to 12 in., thus providing handling distances of about 2 to 10 in. Scissors-type tongs (Fig. 17-1b and c) provide a firm, secure grasp. Depending on type and size of tongs, objects up to about 4 in. in diameter can be picked up. Distance achieved is 4 to 16 in. A 12-in. forceps with a shield of sheet lead (Fig. 17-2) is designed primarily for handling radium and radon needles.

17-8 EQUIPMENT FOR HANDLING, STORAGE, AND TRANSPORTATION

Handling tongs of various types have been designed especially for handling radioactive materials. A high degree of dependability, ease of decontamination, ease of operation, and versatility in size and shape of object handled are important criteria.

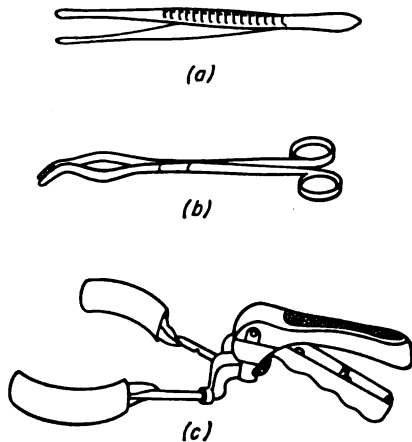


FIG. 17-1. Standard laboratory forceps and tongs. (Fisher Scientific Company.)

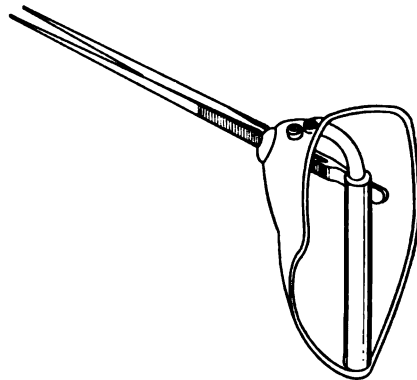


FIG. 17-2. 12-in. forceps with lead hand shield. (Radium Chemical Co., Inc.)

Although the heads are often a finger type (Fig. 17-3) and can grasp objects of many sizes, some heads employ interchangeable (or plug-in) fingers for different purposes. On other tongs, the entire heads are interchangeable. One adaptation employs a noose instead of mechanical fingers. Depending on the individual design, the tongs are made in lengths from $1\frac{1}{2}$ to 10 or 15 ft. Tongs are made to be spring-loaded with the fingers in either a normally open or a normally closed position. Some tongs have the capability of being securely locked on the object grasped. (Additional information on handling tongs can be found in the following references: Selected Reference Material; Chemical Processing and Equipment, *AEC Rept. TID-5276*.

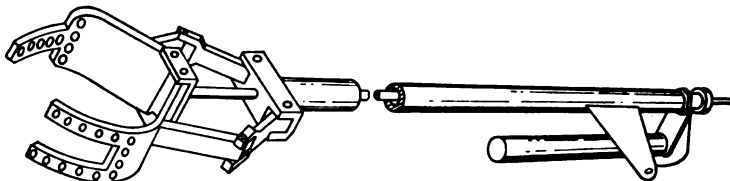


FIG. 17-3. Tongs for handling radioactive material. (Stephenson, R., "Introduction to Nuclear Engineering," p. 355, McGraw-Hill, New York, 1954.)

Fourth Annual Symposium on Hot Laboratories and Equipment, held in Washington, D.C., Sept. 29 and 30, 1955, *AEC Rept. TID-5280*, 1955. Tompkins, P. C., A. Broido, and J. D. Teresi, The Handling of Radioactive Materials in the Experimental Biology Section, *AEC Rept. MDDC-377*, September, 1946. Bizzell, O. M., Equipment for Radioisotope Laboratories, *AEC Rept. AECU-2875*, July, 1954. Apperly, A., Remote Handling Equipment, *Atomic Energy Research Estab. G. B.*

Rept. AERE E/R 1291, 1953. Farmakes, J. R., Snare Type Handling Device, *Nucleonics*, vol. 10, no. 11, p. 90, November, 1952.)

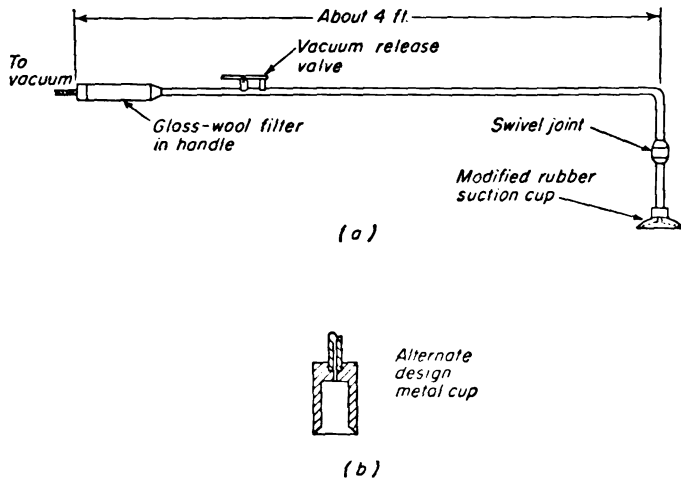


FIG. 17-4. Vacuum-type transfer tongs. (Bizzell, O. M., *Equipment for Radioisotope Laboratories*, AEC Rept. AECU-2875, p. 8, July, 1954.)

Magnetic handlers utilize magnetic heads fitted with extension handles to transfer and manipulate magnetic solids. Either a-c or d-c electromagnets may be used. One design (Tracerlab, Inc.) for small solids uses as the magnetic source a permanent magnet that can be retracted. Cobalt metal, being magnetic, is readily susceptible to such manipulation. In other cases magnetic containers may be used. The repeated use of magnetic handlers must consider the possibility and consequences of magnetizing the objects handled.

Pneumatic gripping devices attached to long handles are used for the transfer of solids. These usually require a continuous source of vacuum controlled by a vent valve. The heads may be of a modified suction cup (Fig. 17-4a) for smooth plane-surfaced objects or cup shaped (Fig. 17-4b) for small regular-shaped objects, e.g., sources. In the latter case, the fit between the head and the object can be quite loose, the air flow into the head holding the object in the cup. (Bizzell, O. M., *Equipment for Radioisotope Laboratories*, AEC Rept. AECU-2875, July, 1954. Brown, G. E., Cobalt 60 Range for the Calibration of Dose and Dose Rate Meters, *Natl. Bur. Standards U.S. Rept.* 2091, December, 1952.)

Bottle cappers (Fig. 17-5) permit the removal of screw-type bottle caps without direct contact. A tapered hole in a rubber head is used to grasp and turn the bottle cap. The operation is facilitated if the bottle is placed on a soft rubber sheet to prevent the bottle from turning as the cap is removed.

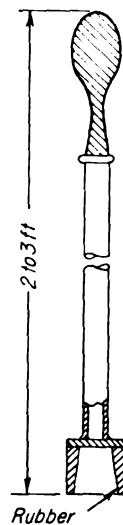


FIG. 17-5. Bottle capper.

REMOTE-CONTROL UNIT-PROCESS EQUIPMENT

When processes must be repeated, the most economic and often the safest approach is to utilize equipment designed for the specific process or the type of process. The use of pressure differentials to transfer liquids (rather than the use of physical manipulation of containers) is often employed in unit process equipment.

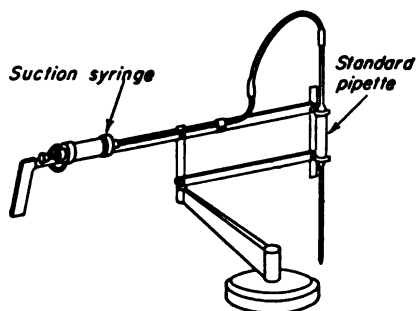


FIG. 17-6. Remote-control pipetting device. (Tracerlab, Inc.)

Remote-control pipetting and solution transfers are accomplished by devices that permit the remote operation and manipulation of pipettes. The device shown (Fig. 17-6) is about 2 ft in length. Clamps secure a standard pipette which can be raised, lowered, or maneuvered over a large area. The pipette is connected to a syringe or pipette control by a flexible tube. A hand-controlled pump or syringe draws the liquid into or expels it from the pipette. Coarse and

fine controls on the syringe permit a wide range of pipette sizes to be used. Quantitative control is achieved by the use of calibrated pipettes. Such devices are used for direct-viewing operations or over low barriers.

Other pipette controls (Fig. 17-7) employ a positive-displacement system for quantitative liquid transfer. The entire system including the pipette is filled with

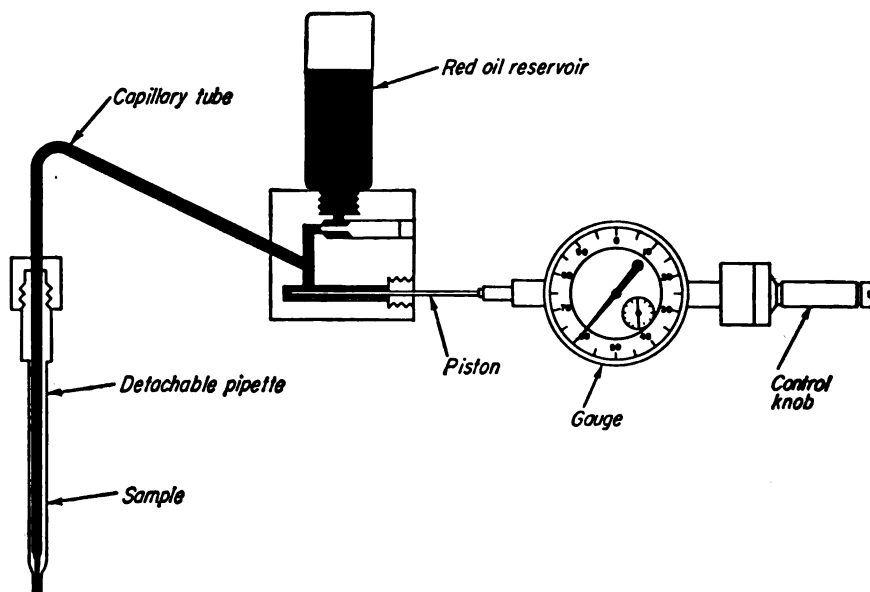


FIG. 17-7. Positive-displacement remote liquid sampler. (Selected Reference Material: *Chemical Processing and Equipment*, AEC Rept. TID-5276, p. 169.)

oil or other liquid. Withdrawing the piston a distance indicated on the gauge allows a like volume of solution to be drawn into the pipette. The control device can be mounted on various types of manipulators that allow it to be maneuvered remotely.

The **Stang precipitator** (Fig. 17-8) uses a vacuum source to control precipitation-filtration operations. The main reaction vessel (normally glass for laboratory processes) has a porous-glass filter disk sealed in the lower portion. The application of a vacuum in the container over the disk will prevent liquid from passing through

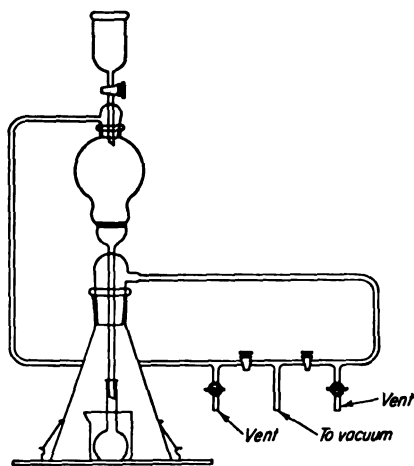


Fig. 17-8. Stang precipitator. (Bizzell, O. M., *Equipment for Radioisotope Laboratories*, AEC Rept. AECU-2875, p. 13, July, 1954.)

the disk. A larger pressure differential will bubble air through the disk to stir the solution. Vacuum applied to the lower side will draw the solution through the filter. In some cases, air pressure is used to control the disk. The basic principle can be applied to many types of equipment.

(Descriptions of **unit-process equipment** for direct-viewing techniques are included in references: Bizzell, O. M., *Equipment for Radioisotope Laboratories*, AEC Rept. AECU-2875, July, 1954. Tompkins, E. R., *Equipment for Radiochemical Operations*, AEC Isotopes Div. Circ. A-6, June, 1948. Shamos, M. H., and S. G. Roth, "Industrial and Safety Problems of Nuclear Technology," Harper, New York, 1950.)

CLOSE SHIELDS

Radiation shields that fit snugly around the container of radioactive material provide protection utilizing a minimum of shielding material and a minimum loss of visibility of the over-all operation.

Lead, concrete, or steel "pigs" (Fig. 17-9) are cylinders with or without a cover that closely fits bottles or other containers. Besides providing shielding, the pigs provide stability that prevents upsetting the container. Soft rubber disks are often

17-12 EQUIPMENT FOR HANDLING, STORAGE, AND TRANSPORTATION

placed in the pigs to grip the bottle as caps are unscrewed. A series of holes in a lead brick can accommodate several bottles.

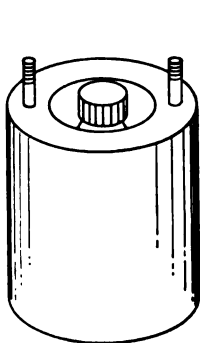


FIG. 17-9. Lead shield. (Ward, D. R., *Design of Laboratories for Safe Use of Radioisotopes*, AEC Rept. AECU-2226, p. 16, November, 1952.)

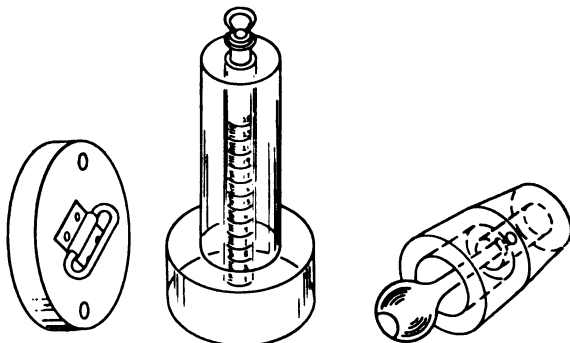


FIG. 17-10. Typical transparent plastic shields. (Tompkins, P. C., et al., *The Handling of Radioactive Materials in the Experimental Biology Section*, AEC Rept. MDDC-377, p. 26, January, 1946.)

Clear plastic shields (Fig. 17-10) are designed to fit closely around standard or special glassware. Such shields also increase the stability. In some cases, e.g., up to 10 mc of P-32, the shielding is sufficient to allow the shields to be handled safely by hand. Such shields have been used with beakers, flasks, centrifuge tubes, serum tubes, and similar items. (Tompkins, P. C., A. Broido, and J. D. Teresi, *The Handling of Radioactive Materials in the Experimental Biology Section*, AEC Rept. MDDC-337, September, 1946. Bizzell, O. M., *Equipment for Radioisotope Laboratories*, AEC Rept. AECU-2875, July, 1954.)

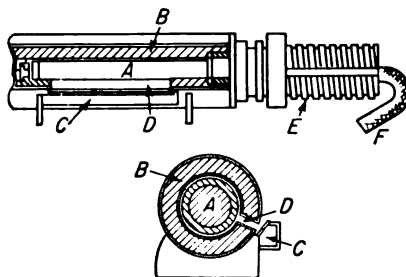


FIG. 17-11. Lead-shielded syringe. (A) Standard 50-cc syringe. (B) Lead shielding. (C) Mirror. (D) Narrow slit in shielding. (E) Operating mechanism. (F) Cord to lamp. (*Nucleonics*, vol. 10, no. 3, p. 25, 1952.)

Shielded syringes are used for the injection of radioactive solutions. For use with beta emitters, the shields are glass or clear plastic. The transparency allows the calibrations on the syringe to be read in the usual manner. Lead shields have been designed for use with gamma emitters. Such shields may have a narrow slit and a mirror (Fig. 17-11) to facilitate reading the scale. Another type (The Hamilton

Company, Whittier, Calif.) has scale marks on the outside of the shield with a pointer attached to the plunger to indicate the readings. Lead-shielded syringes with $\frac{1}{2}$ in. Pb shielding weigh about 4 lb for a 10-cc size and $1\frac{1}{2}$ lb for a 2-cc size.

CONTAMINATION-CONTROL EQUIPMENT

Radiochemical fume hoods (Fig. 17-12) are used to prevent the spread of radioactive contamination from the process material to the laboratory. Such hoods are designed to prevent a minimum of recirculation of hood air into the laboratory

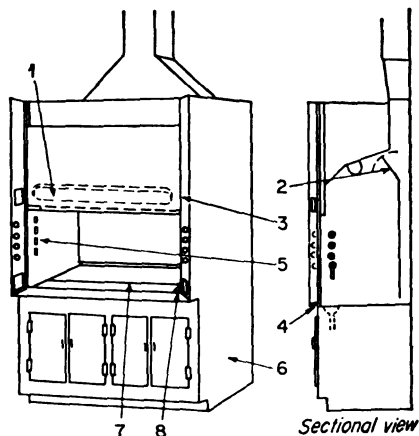


FIG. 17-12. Radiochemical fume hood. (1) The lintel behind the sash contains an air bypass opening which becomes exposed when the sash is lowered. This bypass prevents excessive velocity of air at the work surface when the sash is nearly closed. (2) The withdrawal of both light and heavy fumes may be achieved by the proper adjustment of a movable section of the back baffle. (3) Turbulence of the air entering the hood should be reduced as much as possible. This may be accomplished by use of a 6-in. "picture-frame" air foil at the sides and bottom of the hood. (4) The air-flow characteristics will be improved by the use of a vent space between the bottom air foil and the hood body. (5) Service outlets (gas, water, vacuum, etc., as required) should be located near the front of the hood so the operator will not have to reach into the hazardous zone to make hose connections. (6) The base structure should be strong enough to support a ton of shielding, but it is not recommended that shielding be incorporated in the hood structure. (7) For proper fume control, all operations should be performed beyond a safety line painted 8 in. inside the face of the hood. (8) Easily cleaned cup sinks, located near the front of the hood, are recommended in place of less accessible trough sink located at the back. (Ward, D. R., *Design of Laboratories for Safe Use of Radioisotopes*, AEC Rept. AECU-2226, p. 18, November, 1952.)

spaces. Air velocities of 50 to 80 ft/min are used across the face of the hood. Automatic by-passes are used to prevent air-velocity increases as the door is closed. Exposed surfaces are of a material that is relatively easy to decontaminate. Strip-pable coatings are often used. Such hoods are designed to support the lead walls and barriers. Controls for the services are in accessible locations. The exhaust air is usually filtered. The use of radiochemical fume hoods is based upon the "dilute and disperse" approach to contamination control. Auxiliary contamination-control

17-14 EQUIPMENT FOR HANDLING, STORAGE, AND TRANSPORTATION

ventilation in the form of a small hood, e.g., an inverted funnel, and a trap and/or filter connected to a vacuum supply is often used in hoods for operations likely to generate air-borne contaminant.

Vacuum-bench enclosure (Fig. 17-13) provides a housing for glass reaction trains. Such apparatus is usually sealed and contamination control is good, but it is sus-

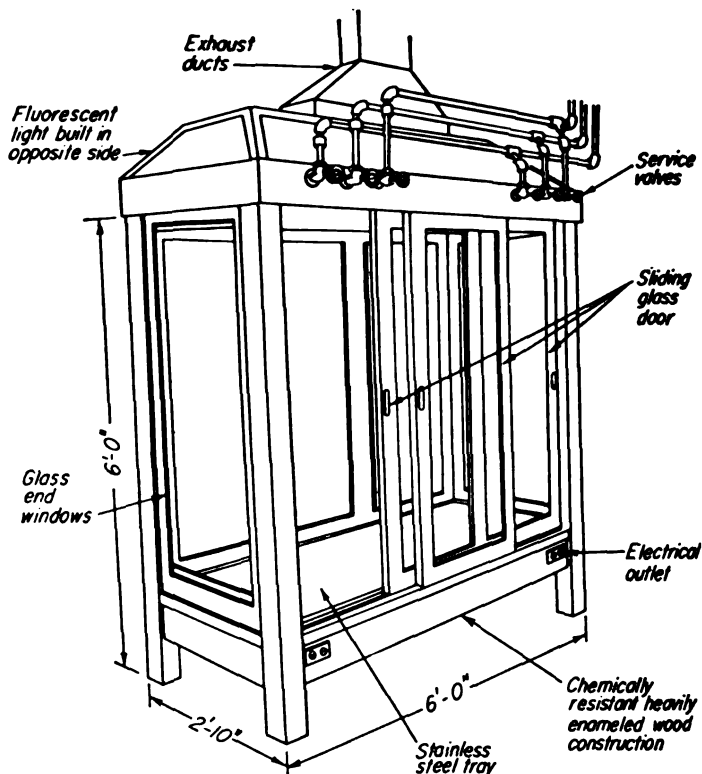


FIG. 17-13. Vacuum-bench enclosure. (Ward, D. R., *Design of Laboratories for Safe Use of Radioisotopes*, AEC Rept. AECU-2226, p. 24, November, 1952.)

ceptible to breakage. The enclosure contains any released material and exhausts it to the atmosphere. Operations in such an enclosure are susceptible to distance control and the use of local shielding.

Gloved boxes (Fig. 17-14) (or "dry" boxes) are used to confine and control contamination. They are total enclosures, the working area being physically isolated from the operator. Exhaust fans provide a negative pressure within the box to control the escape of air-borne contaminant. The small amount of exhaust air is filtered to remove essentially all contaminant. Long gloves sealed to the box allow the operator to manipulate equipment in the box. Gloved boxes are used in this manner for alpha and low-radiation-level work. The use of close shields and tongs in the boxes allows higher radiation levels to be used. Tongs may be mounted in the glove ports to increase the distance further.

Gloved boxes for most work are of a nontight construction, the passage of air into the box through cracks being sufficient to prevent escape of the material. Gloved boxes for work with materials of very high toxicity are designed to be quite tight and incorporate complicated systems to transfer material safely in and out of the box. Several gloved boxes are often joined together in multiple units so that a number of

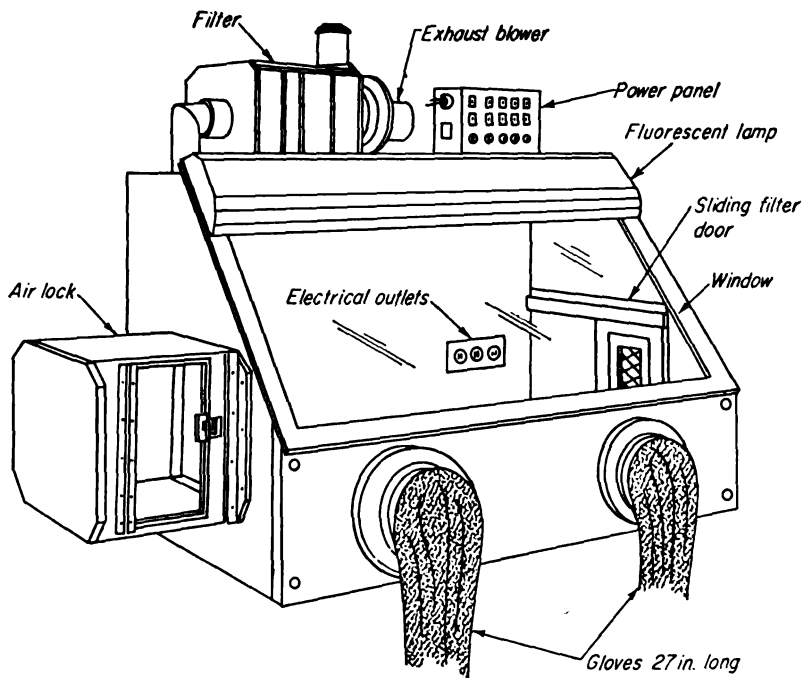


FIG. 17-14. Basic gloved box. (Scientific Service, Inc.)

operations can be performed without removing the process material from the controlled atmosphere. Such boxes are often ventilated with an inert gas to combat the pyrophoric nature of some materials. Gloved boxes of this type are described in detail by Kelman, L. R., *et al.*, *Nucleonics*, vol. 14, nos. 3, 4, 5.

SHIELDING-BARRIER EQUIPMENT

Barriers are used to shield occupied areas from the radiation from radioactive materials in operating areas in cases:

1. When direct-viewing techniques, especially local shielding, cannot be employed economically or practically
2. When a versatile intermediate-level facility (i.e., permanent or semipermanent setup) is desired for a wide variety of operations over a period of time

Such barriers are used when radiation levels do not require consideration of gamma scattering from adjoining walls or other solids. Consequently, the top and perhaps one or more sides of the process area are not shielded. Such barriers are

17-16 EQUIPMENT FOR HANDLING, STORAGE, AND TRANSPORTATION

usually made of opaque solid materials. Consequently, both the viewing and the manipulative techniques require penetration or circumvention of the barrier. In the case of viewing, this leads to the use of equipment in which distortion, reversed images, lack of depth perception, etc., are problems. In the case of manipulation, mechanisms become more complex and heavier, force reflectance becomes more difficult, and dexterity is generally reduced.

If the contamination potential is low, such operations can be set up in the open. Otherwise, the use of a ventilating hood or fume hoods is required. Special barriers for gloved boxes and special fume hoods are used in some cases where contamination control is critical.

Reach-around barricades such as the "L block" (Fig. 17-15) are often used for the preparation and dismantling of radium applicators or similar devices. The body

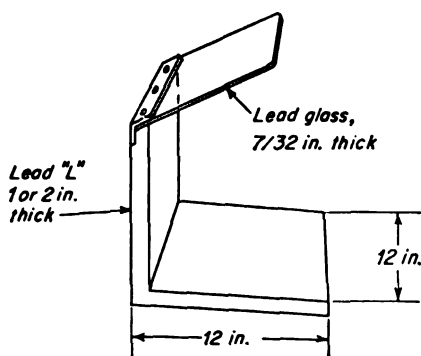


FIG. 17-15. Reach-around barricade. (Atomic Energy of Canada, Ltd.)

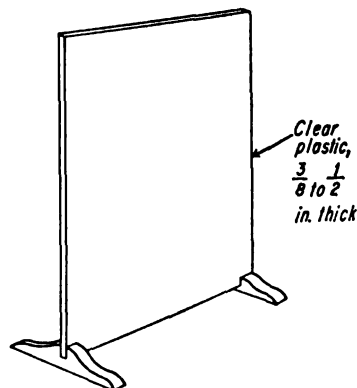


FIG. 17-16. Beta barrier. (Bizzell, O. M., *Equipment for Radioisotope Laboratories, AEC Rept. AECU-2875*, p. 5, July, 1954.)

of the operator is protected by 1- or 2-in. thickness of lead. NBS Handbook 54 recommends that the shield be set back 12 in. from the edge of the table to help maintain a safe distance. Operations are viewed through a high-density "visor." Work is performed with forceps. Shielded storage facilities are located within reach. Such facilities are recommended for a maximum of 160 mc-hr of radium when tongs are used to keep the hands 12 in. from the source. ("Protection against Radiation from Radium, Cobalt-60 and Cesium-137," NBS Handbook 54, September, 1954.)

Beta barriers for the shielding of operations involving energetic beta emitters are often transparent plastic sheets, $\frac{3}{8}$ to $\frac{1}{2}$ in. thick, mounted on feet for stability (Fig. 17-16). Shields may be rather narrow to facilitate use of tongs around sides; wider shields may have ports (with covers) for the insertion of tongs or other manipulators. These barriers are used for the 10- to 100-mc range of the higher-energy beta emitters. These barriers may be used in the open or in radiochemical fume hoods, depending upon the contamination potential. (Bizzell, O. M., *Equipment for Radioisotope Laboratories, AEC Rept. AECU-2875*, July, 1954.)

Brick barricades of lead, steel, or other materials are used when the gamma radiation level becomes too high for direct-viewing techniques. A typical temporary setup is shown in Fig. 17-17. The wall is constructed high enough to keep the

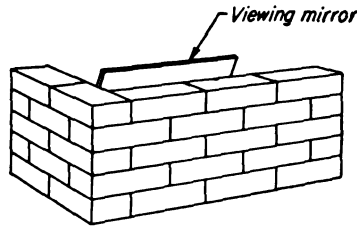


FIG. 17-17. Brick barricade.

worker out of the direct beam of the source but low enough to utilize a simple mirror setup. Manipulations are performed with straight tongs (Fig. 17-3) or with tongs that have one or more right angles in the extension shaft. Remote pipetters (Fig. 17-6) can be used with low barriers. Lead bricks are usually a nominal size of 4 by 8 by 2 in. They are available in various shapes (Fig. 17-18) that permit interlocking for radiation "tightness" and improved stability of walls.

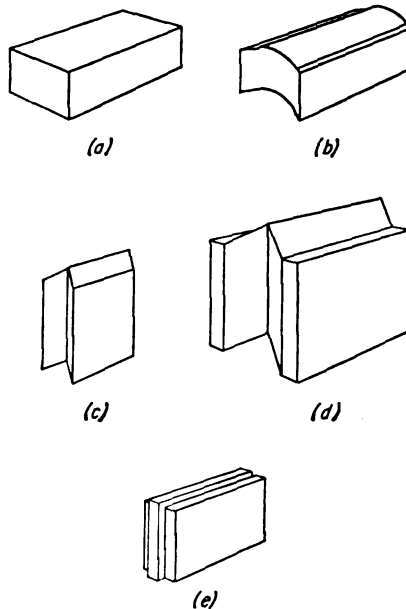


FIG. 17-18. Commercial lead bricks. (a) Butt, 4 by 8 by 2 in. Available in other sizes; also available in steel, iron, marble, and framed high-density glass. (b) Curved, 4 by 8 by 2 in. Available in other sizes. (c) Saw-tooth, 2 by 4 by 4 in. (d) Saw-tooth, 6 by 8 by 6 in. Also available in various sizes in high-density concrete. (e) Tongue and groove, $1\frac{1}{2}$ by 6 by 3 in. (*Nucleonics*, vol. 13, no. 5, p. 84, 1965.)

17-18 EQUIPMENT FOR HANDLING, STORAGE, AND TRANSPORTATION

Permanent or semipermanent **caves** (Fig. 17-19) provide working areas for handling curie quantities of gamma emitters. Caves are constructed of concrete, lead, or steel in either brick or a monolithic form. For shielding purposes, the walls extend 6 to 8 ft in height and may be long enough to accommodate several experimental setups. Caves are usually ventilated for a "dilute and disperse" contamination control. Viewing must be "over-the-top" or penetrate the wall.

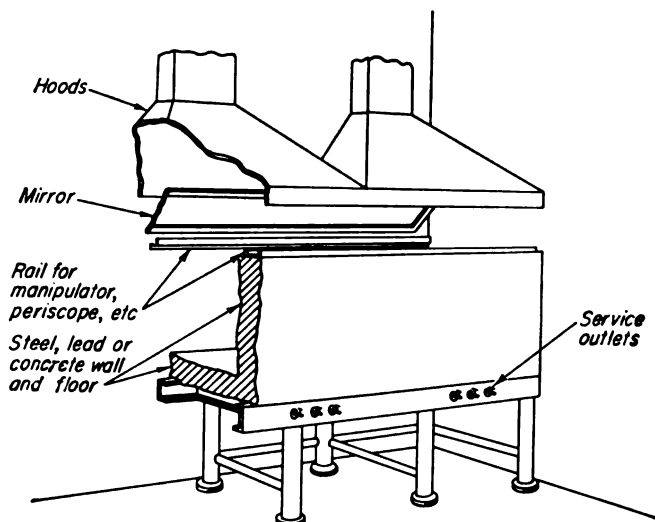


FIG. 17-19. Essential features of basic cave.

Mirror systems are often used for viewing. **Over-the-wall periscopes** (Fig. 17-20) permit close-up viewing. Such periscopes can be mounted on tracks on the top of the wall to facilitate traverse of the area. Other periscopes are built-in or inserted in the wall (Fig. 17-21). Prisms are used to obtain a Z-shaped viewing passage through the wall. The position of inner prism is adjustable to increase the vertical viewing range. The periscope can be used with a mirror in the cave to view along the length of the cave. This mirror can be rotated on its vertical axis by a motor drive to increase the viewing area. Windows of high-density glass or hollow windows filled with water, mineral oil, or other solutions are often used in caves.

Manipulations are performed by tongs mounted in ball joints (Fig. 17-22) that are mounted in the wall itself. Such tongs often have interchangeable heads for performing various operations. Rectilinear manual manipulators (Fig. 17-23) designed for over-the-wall use ride on tracks on the top of the wall. Master-slave manipulators are also used in some cases.

Special-purpose **remote-control equipment** such as pipetters, exposure can openers, metallographic equipment, testing, and metalworking equipment are often used in caves. (For additional information see references: Selected Reference Material; Chemical Processing and Equipment, *AEC Rept. TID-5276*. Hot Labs—A Special Report, *Nucleonics*, vol. 12, no. 11, pp. 35-100, November, 1954. Fourth Annual Symposium on Hot Laboratories and Equipment, held in Washington, D.C.,

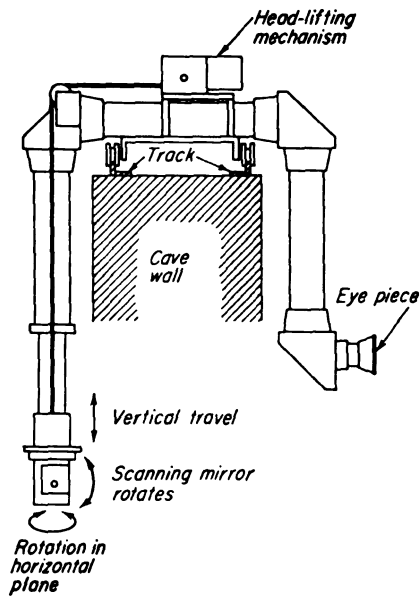


FIG. 17-20. Oak Ridge National Laboratory over-the-wall periscope, motorized. (Selected Reference Material; Chemical Processing and Equipment, AEC Rept. TID-5276.)

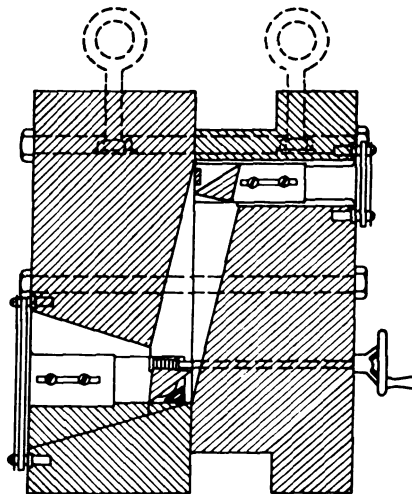


FIG. 17-21. Through-wall periscope. (Nucleonics, vol. 12, no. 11, p. 39, 1954.)

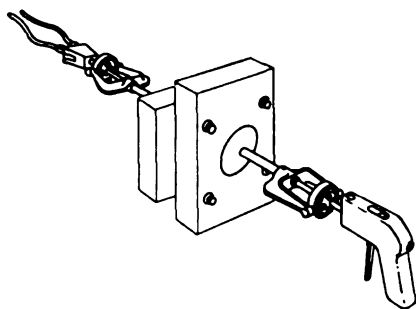


FIG. 17-22. Ball-joint tongs. Ball-type handler consists of tube through a ball mounted in two cast-lead forms. For sensitivity, force at trigger has 1:1 ratio with that at interchangeable tongs. Ball is 9-in. hollow cast-iron lead-filled sphere drilled on diameter and bushed with an oilite bearing; hole is reamed to sliding fit with rod. Ball is supported in lead form with concave surface whose curvature is that of ball radius plus diameter of 1/8-in. ball bearings. (*Nucleonics*, vol. 12, no. 11, p. 39, 1954.)

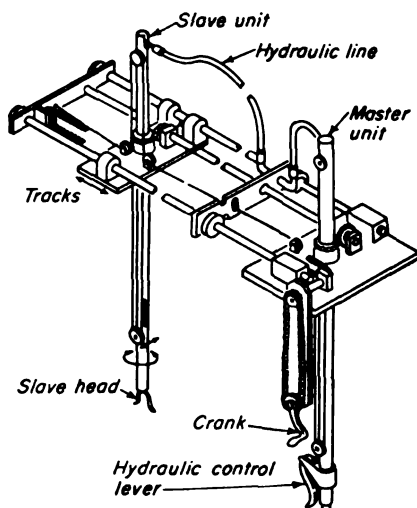


FIG. 17-23. Rectilinear manipulators, Brookhaven National Laboratory model 2. (*Selected Reference Material; Chemical Processing and Equipment*, AEC Rept. TID-5276, p. 135.)

Sept. 29 and 30, 1955, *AEC Rept. TID-5280*, 1955. Fifth Hot Laboratories and Equipment Conference 1957 Nuclear Congress, Philadelphia, March, 1957.)

Mobile shields (Fig. 17-24) are used to approach objects, e.g., cyclotron targets, that are difficult to shield by other techniques. A viewing window and tongs allow manipulative work to be performed. Similar units have been constructed on self-propelled units, thus facilitating the mobility and allowing greater thicknesses of

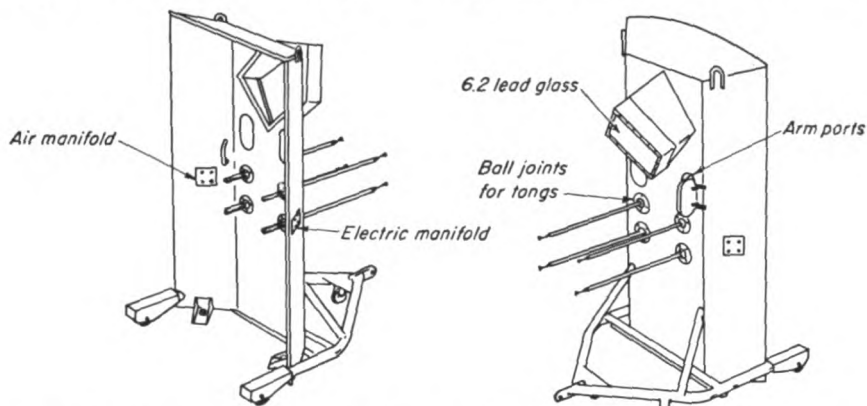


FIG. 17-24. Mobile personnel shield. (*Fourth Annual Symposium on Hot Laboratories*, *AEC Rept. TID-5280*, p. 196, 1955.)

shielding to be used. (Hot Labs—A Special Report, *Nucleonics*, vol. 12, no. 11, pp. 35-100, November, 1954. Fourth Annual Symposium on Hot Laboratories and Equipment, held in Washington, D.C., Sept. 29 and 30, 1955, *AEC Rept. TID-5280*, 1955.)

"Gloved" box shields (Fig. 17-25) are used to enclose gloved boxes when rigid contamination control as well as radiation-dosage control is required. The shield

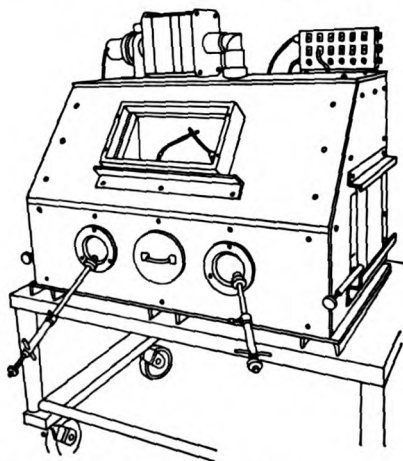


FIG. 17-25. 1-in. lead shield for gloved box. (*Scientific Service, Inc.*)

17-22 EQUIPMENT FOR HANDLING, STORAGE, AND TRANSPORTATION

usually encloses five or six sides of the box. The gloves are replaced by tong manipulators in ball joints. Special-purpose remote-control equipment is often used. The shields are 1-, 2-, or 4-in.-thick lead. Windows are either laminated or solid high-density glass. The gloved box can be removed from the shield and other boxes inserted. Gloved-box shields are usually mounted on wheeled dollies.

Portable shielding for fume hoods (Fig. 17-26) provides for the intermittent use of radiochemical fume hoods at intermediate radiation levels. The hoods are

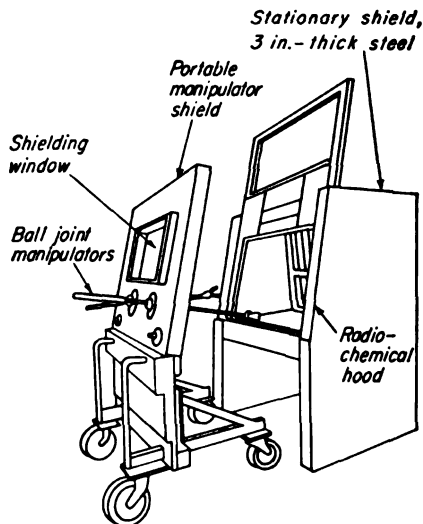


FIG. 17-26. Portable shield for fume hood, Argonne National Laboratory. (*Selected Reference Material; Chemical Processing and Equipment, AEC Rept. TID-5276, p. 63.*)

permanently shielded on the sides, bottom, and back with steel plate. The portable shield can be moved over the front of the hood when required. A high-density window is provided. Manipulation is accomplished by tongs in ball joints. Other ports are provided for access. (*Selected Reference Material; Chemical Processing and Equipment, AEC Rept. TID-5276.*)

Junior caves (Fig. 17-27) provide permanent but movable front shields for special radiochemical fume hoods. The hoods are designed specially to provide for access of master-slave manipulators, access doors in the front panel, transfer chambers, etc. Special ventilation involves introduction in one side of box and exhaust out the other at a rate of 200 to 300 cu ft/min. Shielding thickness is about 3 in. of steel on all sides. The front panel can be moved to the side to permit limited access. Ball-joint manipulators are used as well as the master-slave manipulators. (*Selected Reference Material; Chemical Processing and Equipment, AEC Rept. TID-5276.*)

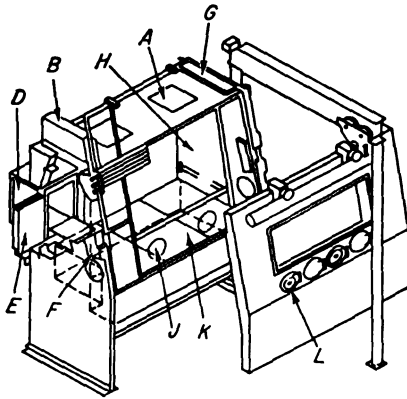


FIG. 17-27. Cutaway view of junior cave, Argonne National Laboratory, model 1. (A) Master-slave manipulator port. (B) Air inlet. (D) Hinged door. (E) Sliding door. (F) Utility access. (G) Exhaust-duct connection. (H) Prefilter (inside). (J) Glove port. (K) Removable tray (inside). (L) Manipulator ports. (*Selected Reference Material; Chemical Processing and Equipment, AEC Rept. TID-5276, p. 69.*)

CLOSED-CELL EQUIPMENT

When quantities of gamma-emitting radioactive materials become large enough, e.g., about 10 to 20 curies of I-131 or about twice that of a higher-energy emitter such as Co-60, and continuous use is anticipated, scattered radiation dictates the use of totally enclosed cells. Because of the inherent massiveness required, such cells are usually an integral part of the building construction.

Hot cells have been designed for the handling of megacurie quantities of 1-Mev gamma emitters, thus resulting in walls up to about 5 ft thick. Contamination control is maintained by ventilating the cell and filtering the exhaust air. When alpha-emitting materials are present in addition to the beta-gamma emitters, additional precautions must be taken to prevent air leakage or loss. The transfer of contaminated objects must be carefully considered and closely controlled. Tighter fits or sealed openings reduce the volume of contaminated air that must be cleaned before exhausting. (The following references describe various **hot-cell designs** in detail: Hot Labs—A Special Report, *Nucleonics*, vol. 12, no. 11, pp. 35-100, November, 1954. Fourth Annual Symposium on Hot Laboratories and Equipment, held in Washington, D.C., Sept. 29 and 30, 1955, *AEC Rept. TID-5280*, 1955. Fifth Hot Laboratories and Equipment Conference, 1957 Nuclear Congress, Philadelphia, March, 1957. Rupp, A. F., Methods of Handling Multikilocurie Quantities of Radioactive Materials, *Proc. Intern. Conf. Peaceful Uses Atomic Energy, Geneva*, vol. 14, pp. 128-155, 1956. Yecker, D. F., Hot Laboratories, *Reactor Handbook*, vol. 2, Engineering, chap. 7.3, pp. 909-923, *AEC Rept. AECD-3646*, May, 1955.)

The University of California Radiation Laboratory employs a system considerably different from the usual hot cell for high alpha-beta-gamma work. Airtight steel cabinets similar to "gloved" boxes are used with thick lead shields and "castle" tongs. In order to reduce the concentration of air-borne contaminants to

17-24 EQUIPMENT FOR HANDLING, STORAGE, AND TRANSPORTATION

acceptable levels, the air in the cabinet is circulated through a blower-scrubber-condenser-filter system at 10 cu ft/min and returned to the cabinet. A negative pressure in the cabinet is maintained by a separate blower-filter system that discharges only 0.02 cu ft/min to a stack. (Fourth annual Symposium on Hot Laboratories and Equipment, held in Washington, D.C., Sept. 29 and 30, 1955, *AEC Rept. TID-5280*, pp. 338-346, 1955.)

The more common nonproduction hot cells can be roughly classified as **manipulator cells** and **process cells**.

Figure 17-28 shows the cross section of a typical high-level **manipulator cell**. Interior dimensions are about 6 by 6 by 13 ft. At some installations, such cells are built in rows in order that common shielding is used between cells. The transfer of

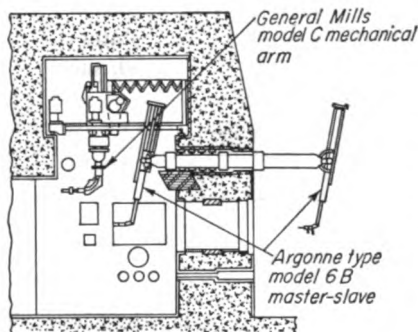


FIG. 17-28. High-level manipulator cell. (*Nucleonics*, vol. 12, no. 11, p. 43, 1954.)

radioactive material from one cell to another is facilitated by such an arrangement. Viewing is accomplished through dense windows. Primary manipulation is accomplished by master-slave manipulators although heavy-duty manipulators or remote-operated cranes are often used.

Process cells are used primarily for chemical processing of liquids. Transfers are accomplished principally by the application of pressure differentials. Mechanical manipulations are ones that are simply adapted remote-control operation by reach rods and flexible remote-

control cables, e.g., valve operation. Consequently, manipulators are seldom used. In many setups visual requirements are low or nonexistent; consequently, periscopes are often the only viewing devices used.

Cells vary considerably in detail from installation to installation. Subsequent sections attempt to outline some of the requirements and specifications for different components that are being used.

Closed-cell **construction materials** must provide satisfactory radiation shielding. Table 17-1 lists materials that are being used in various installations.

Often a combination of materials are used, i.e., high-density concrete for the operating face where the greatest attenuation is required and regular concrete for other walls and the roof.

The **interior surfaces** of the cells are designed for ease of decontamination. Concrete surfaces are painted. Steel liners that also serve as the forms for the concrete are often used. Stainless-steel and lead liners have also been employed. Regardless of the lining material, surfaces are usually painted with corrosion-resistant paints to facilitate decontamination. Additional coatings that are strippable are often used.

Most work in cells requires that at least some phase of the operation be visually inspected. The most common viewing devices used are **shielding windows** of three types: the liquid-filled (usually ZnBr_2 solution), high-density glass, and the composite high-density glass and zinc bromide solution. Table 17-2 lists the properties of the basic materials. (More complete details on **viewing devices** are included in

Table 17-1. Materials for Gamma Shielding *

Type of shield	Sp. gr., g/cc	Density, lb/cu ft
Concretes:		
Limestone-plasterer's sand	2.2	140
Gravel-sand	2.4	152
Barytes-barytes sand	3.4	215
Magnetite-magnetite fines	3.65	228
Barytes-iron shot	4.15-4.30	260-268
Ferrophosphorous-ferrophosphorous fines	4.7	293
Metallic iron-iron shot	5.9-6.18	370-385
Steel	7.8	490
Metallic lead	11.3	706

* Glen, H. M., Materials of Biological Shielding, 2d Nuclear Engineering and Science Conference, Paper 57-NESC-51, ASME, New York, 1957. Costs as well as other pertinent information on structural shields are included in Glen as well as: Stephenson, R., "Introduction to Nuclear Engineering," p. 248, McGraw-Hill, New York, 1954. Davis, H. S., How to Choose and Place Mixes for High Density Concrete Reactor Shields, *Nucleonics*, vol. 13, no. 6, pp. 60-65, June, 1955. Narver, David L., Jr., "High Unit Weight Concretes for Radiation Shielding," Holmes and Narver, Inc., Los Angeles, Calif.

Table 17-2. Transparent Shielding Materials

Type	Manu- facturer's no.	Density, gm/cc	Index of refrac- tion	Color	Manu- facturer and/or ref.
Commercial lime		2.52	1.52	Green	Pittsburgh *
Water-white lime		2.52	1.52	Colorless	Pittsburgh *
Nonbrowning lime	8365	2.67	1.52	Water-white	Corning
Nonbrowning lead	8362	3.27	1.59	Practically water- white	Corning
Nonbrowning lead	ME-D4.0 NB	4.05	1.66	Yellow	Penberthy
Medium-density lead	ME-D4.8 NB	4.8	1.76	Pale yellow	Penberthy
X-ray lead		4.88	1.76	Deep yellow	Pittsburgh *
High-density lead	HI-D 6.2	6.20	1.97	Yellow	Penberthy
High-density lead	8363	6.22	1.98	Straw	Corning
Zinc bromide solution		2.52	1.56	Colorless	*

* Monk, G. S., K. R. Ferguson, and D. F. Yecker, Remote Viewing, in "Reactor Handbook, vol. 2, Engineering," chap. 7.2, pp. 785-908; *AEC Rept.* AECD-3646, May, 1955.

17-26 EQUIPMENT FOR HANDLING, STORAGE, AND TRANSPORTATION

the following references: Selected Reference Material; Chemical Processing and Equipment, AEC Rept. TID-5276. Fourth Annual Symposium on Hot Laboratories and Equipment, held in Washington, D.C., Sept. 29 and 30, 1955, AEC Rept. TID-5280, 1955. Monk, G. S., K. R. Ferguson, and D. F. Yecker, Remote Viewing, in "Reactor Handbook: Engineering," chap. 7.2, pp. 785-908; AEC Rept. AECD-3646, May, 1955.)

Liquid windows (Fig. 17-29) are usually removable tanks that fit in openings in the cell wall. The cover plates are laminated glass 1 to 2 in. thick. Because of the exposure to radiation, the glass facing the interior is cesium-stabilized nonbrowning

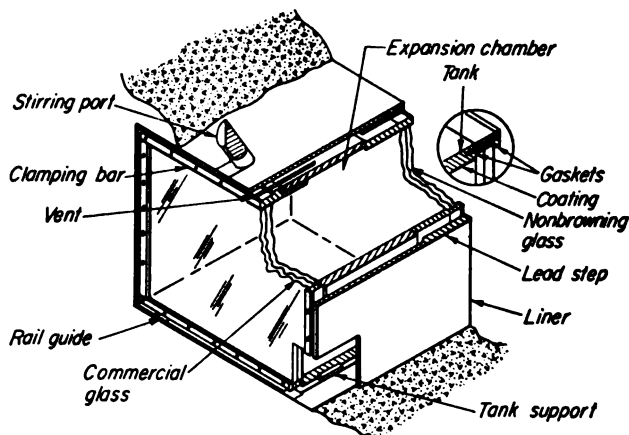


FIG. 17-29. Liquid-filled shielding window. (Selected Reference Material; Chemical Processing and Equipment, AEC Rept. TID-5276, p. 95.)

glass. It is desirable that the window have the same shielding effectiveness as the wall in which it is installed. Consequently, zinc bromide solution ($\rho = 2.5$) is commonly used in conjunction with normal-density concrete walls. Zinc bromide solutions are somewhat unstable, resulting in discoloration of the material. The use of stabilizing agents, the careful consideration of construction materials, and improvement of the quality of the material have eliminated many of the above operational problems. It is considered that exposure to about 10^7 r can be tolerated before remedial action is required. A maximum permissible intensity is about 5×10^4 r/hr because of radiation-induced gas generation in the liquid. This in turn limits the thickness of all-liquid windows to about $4\frac{1}{2}$ ft.

Glass windows (Fig. 17-30) are used in installations where the shielding required is that equivalent to up to 6 ft of normal-density concrete. At higher radiation intensities, radiation-induced discoloration and heating can become problems. The variety of glasses available allows windows to be constructed with shielding effectiveness to thickness ratios equivalent to concrete, high-density concretes, and iron or steel. Most glasses are subject to discoloration by radiation exposure. The application of heat and/or high-intensity white light partially relieves the discoloration. Nonbrowning glasses which are cesium-stabilized are considerably more resistant to radiation discoloration. Although exposure to environmental

conditions causes considerable variation, the nonbrowning glasses can withstand exposures of about 10^8 r before replacement.

Since the maximum thicknesses of large-area windows of high-density glass produced to date are 4 to 6 in., most windows require multiple thicknesses of glass. The inside sheet (i.e., that nearest the radiation source) or sheets should be nonbrowning glass. If lead glass is used, the outside sheet should be safety glass to protect the lead glass from corrosion and scratching. The layers of glass are separated about $\frac{1}{32}$ in. and the space is filled with mineral oil or zinc bromide solution to reduce reflection losses. Thus, the frame is required to be a tank similar to the liquid window tank.

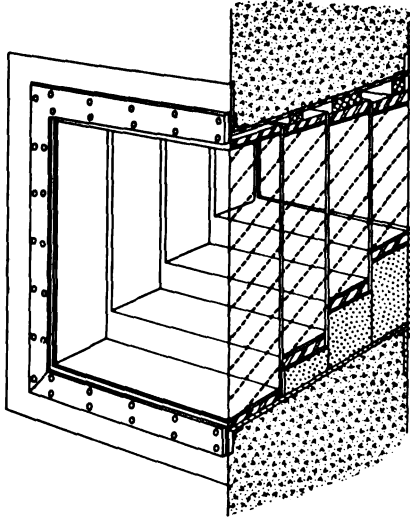


FIG. 17-30. High-density glass window. (Selected Reference Material; Chemical Processing and Equipment, AEC Rept. TID-5276, p. 101.)

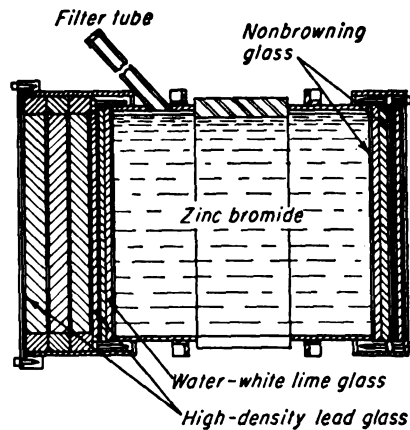


FIG. 17-31. Composite window. (Nucleonics, vol. 12, no. 11, p. 43, 1954.)

Composite windows (Fig. 17-31) are used primarily for installations in high-density concrete where radiation intensities are high. A typical design utilized a laminated section of nonbrowning glass on the inner face. The main section is a zinc bromide window which gives high optical quality. The other layer or layers of lead glass allow reduced thickness.

Periscopes are used in place of windows when physical manipulations are not complicated (e.g., when remotely controlled unit-process equipment is used instead of manipulators). However, in some cases (e.g., for remote microscopy) special periscopes are required in addition to windows.

Figure 17-32 illustrates a **common periscope** and access-hole design. The angle entrance at an interior corner limits radiation escape to scattered radiation. The rotation of a mirror at the objective and coupled with rotation of the periscope itself allows viewing of most of the cell. When higher radiation intensities are anticipated, periscopes with multiple offsets (Fig. 17-33) are used.

17-28 EQUIPMENT FOR HANDLING, STORAGE, AND TRANSPORTATION

Optical systems which present a properly oriented and nondistorted image are complex, particularly if multiple lenses are provided in order to improve the magnification factor. In addition, the design of parts of the periscope exposed to radiation, e.g., lenses, mirrors, etc., must consider any radiation effects. Corrosion resistance and ease of decontamination are also important.

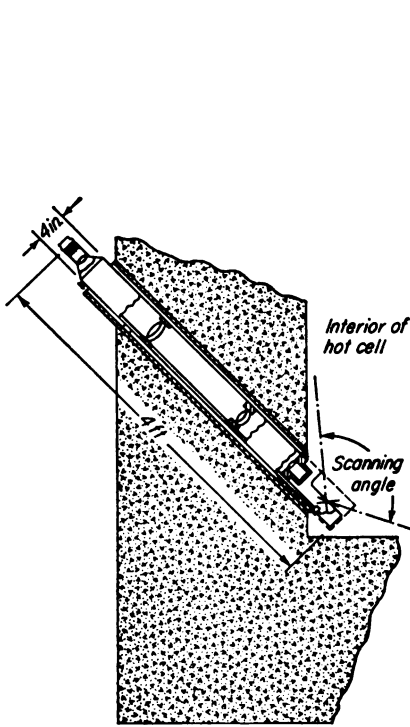


FIG. 17-32. Common periscope. (Stephenson, R., "Introduction to Nuclear Engineering," McGraw-Hill, New York, 1964.)

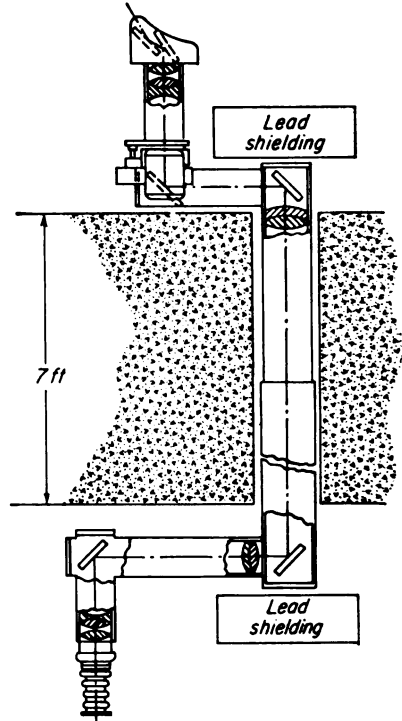


FIG. 17-33. Offset periscope. (Stephenson, R., "Introduction to Nuclear Engineering," McGraw-Hill, New York, 1964.)

Closed-circuit television systems have been used satisfactorily for various remote-viewing requirements, but their application has not become widespread. The lack of depth perception is a serious limitation when television is used for manipulator control. Three-dimensional television systems may ultimately eliminate this difficulty.

Physical **access** to the hot cell is required in order to introduce or remove equipment, radioactive materials, process material, control lines, instrument leads, etc. The design of openings must consider radiation protection as well as contamination control. Most cells provide some sort of door or access hatch. Relatively lightweight doors are often divided and each section mounted on hinges. Heavier doors are mounted on flanged wheels that roll on rails. Others are hung from an overhead rail. Rolling doors are motorized, driven hydraulically, or provided with a winch

or some other device to increase the mechanical advantage. Other doors are raised or lowered hydraulically. At least one installation has the doors split horizontally in the center and the sections linked mechanically to counterbalance each other. Doors are made of solid steel, steel shells filled with concrete or lead, etc.

Access ports are provided in most cells. These are usually cylindrical holes, often stepped to eliminate any gaps. Such holes are stoppered with solid plugs when not in use. Special plugs that have smaller tubes or pipes inset in the shielding material provide access for electrical leads, process tubing, control rods, etc. When practical, the small access tubes are formed in a shallow S shape to eliminate a direct radiation path.

Access hatches, i.e., square or rectangular stepped openings in the top surface of the cell, allow larger objects to be introduced into the cell. Such openings are usually plugged by shields that can be lifted out by a crane.

Special access is often required in special-purpose cells for introduction of radioactive shipments, etc. Such setups often have hoists, propelled dollies on tracks, or similar devices that allow safe handling of standardized items.

Because of light absorption by windows, **cell lighting** requires a relatively high illumination level. Monochromatic sources are required when fine detail must be observed through thick windows. Both sodium-vapor and fluorescent-type mercury-vapor lamps are used for this purpose. Additional incandescent or fluorescent lamp lights are used for general cell illumination. Consideration should be given to the replacement of lamps by remote-control or removable fixtures.

Cells for many uses require "tools" for making physical transfers or repositioning. When a wide variety of different and complex manipulations are required and/or the requirements of the movements cannot be predicted, manipulators are required.

The **mechanical master-slave manipulator** model 8 developed at Argonne National Laboratory (Fig. 17-34) is widely used. In principle, the master-slave manipulator duplicates at the slave end all movements made at the master end. Seven degrees of freedom are required: three in rotation of the head, three in translation, and one of grasping by the tongs. The model 8 manipulator utilizes cables, metal tapes, and gears to connect and transmit the motions. All such connections are contained in the 8-in. tube through the shielding enclosure. Such mechanical linkages are force-reflecting. This gives the operator "feel," thus prevents crushing or other

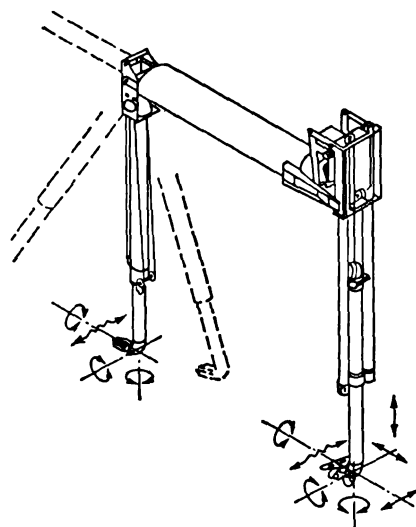


FIG. 17-34. Mechanical master-slave manipulator, Argonne National Laboratory model 8. (*Selected Reference Material: Chemical Processing and Equipment, AEC Rept. TID-5276, p. 123.*)

17-30 EQUIPMENT FOR HANDLING, STORAGE, AND TRANSPORTATION

damage to the objects handled or to the manipulator itself. Each hand can manipulate 5 to 10 lb.

For enclosures that are sealed, **electrically controlled master-slave manipulators** (Fig. 17-35) have been developed. Since electric cables are the only connection between the master and the slave units, greater mobility can be achieved and greater

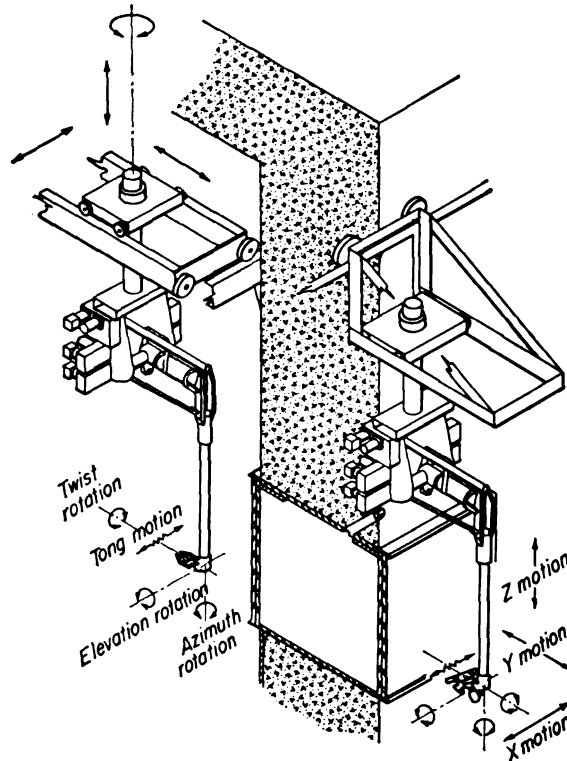


FIG. 17-35. Electrically controlled master-slave manipulators. (*Fourth Annual Symposium on Hot Laboratories and Equipment, held in Washington, Sept. 29 and 30, 1955, AEC Rept. TID-5280, 1955.*)

distances between the units can be achieved. The slave unit can be mounted on bridge crane carriage or a remotely controlled wheeled "robot." A separate force-reflecting servosystem is used to maintain positional synchronism between the master and slave arms in each of the seven degrees of freedom. Radiation exposure is limited to 10^7 to 10^8 r to prevent lubricant and insulation damage. Because of the remoteness and mobility possible by the system, television viewing may be required.

Various types of **electric- or hydraulic-powered manipulators** are used principally for work (i.e., forces, lift capacity, etc.) that requires greater effort than that which can be delivered by the operator through a master-slave system. Most of these units are capable of handling lift loads of up to several hundred pounds. Figure

17-36 shows the **Tell manipulator** which, although light-duty, is typical in its main features of other heavy-duty manipulators. Basic rectilinear motions in three axes are achieved by a bridge-crane arrangement with an additional vertical drive. The manipulative unit consists of "shoulder," "elbow," and "wrist" rotational units. The "hand," or tong, is remotely replaceable for grasping differently shaped objects. Such units are controlled by pistol-grip "joy sticks" and individual switches. Two or three speeds of motion can be obtained. Gripping force is controlled by preadjusting to a designated pressure level or range. Visual means (i.e., meters and audio signals indicating the relative grasping force) are employed on some units. Depending on the purpose of the cell, these units may be used independently or in conjunction with master-slave manipulators. (Additional information on various types of manipulators can be found in the following references: Selected Reference Material; Chemical Processing and Equipment, *AEC Rept. TID-5276*. Fourth Annual Symposium on Hot Laboratories and Equipment, held in Washington, D.C., Sept. 29 and 30, 1955, *AEC Rept. TID-5280*, 1955. Fifth Hot Laboratories and Equipment Conference, 1957, Nuclear Congress, Philadelphia, March, 1957. Hot Labs—A Special Report, *Nucleonics*, vol. 12, no. 11, pp. 35-100, November, 1954. Goertz, R. C., and F. Bevilacqua, Remote Handling, in "Reactor Handbook: Engineering," chap. 7.1, pp. 855-873; *AEC Rept. AECD-3646*, May, 1955. Burnett, J. R., R. C. Goertz, and W. M. Thompson, Mechanical Arms Incorporating a Sense of Feel for Conducting Experiments with Radioactive Materials, *Proc. Intern. Conf. Peaceful Uses Atomic Energy, Geneva*, vol. 14, pp. 116-121, 1956.)

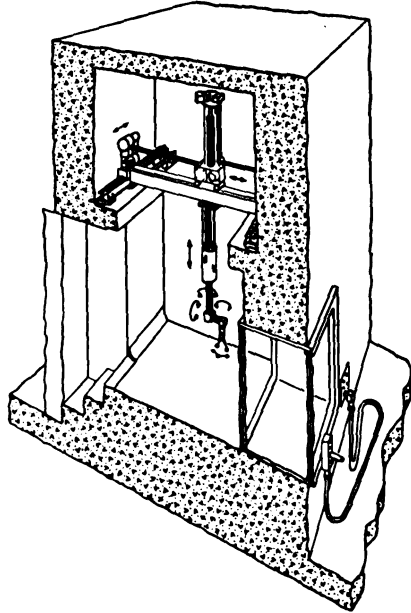


FIG. 17-36. Tell manipulator. (Stephenson, R., "Introduction to Nuclear Engineering," McGraw-Hill, New York, 1954.)

UNDERWATER FACILITIES

Water-filled canals, swimming pools, or pits are used for the handling and storage of insoluble solid or encapsulated radioactive materials. Depths of 8 to 20 ft provide adequate shielding. The pits are lined in some cases with tile or painted to prevent leaching and improve decontaminability. Submarine incandescent lamps are used for illumination. The water should be circulated through filters and ion-exchange columns to provide for decontamination and repurification. pH control is desirable to prevent corrosion of materials (if too acid) and prevent haziness (if too caustic).

17-32 EQUIPMENT FOR HANDLING, STORAGE, AND TRANSPORTATION

Long tongs are used for basic manipulations. Various simple tools such as vises, screw drivers, and hacksaws can be provided with extension handles and used satisfactorily. Many hand-power tools can be used with simple modifications. Remotely controlled special-purpose equipment has been built for continuous immersion. Special periscopes are used for close inspections. (Information on **canals** and the specific equipment used in them is included in the following references: Fourth Annual Symposium on Hot Laboratories and Equipment, held in Washington, D.C., Sept. 29 and 30, 1955, *AEC Rept. TID-5280*, 1955. Rupp, A. F., *Methods of Handling Multikilocurie Quantities of Radioactive Materials*, *Proc. Intern. Conf. Peaceful Uses Atomic Energy, Geneva*, vol. 14, pp. 128-135, 1956.)

STORAGE EQUIPMENT

The storage of radioactive material principally involves shielding the material when it is not being used. The problem can be very simple or very complex, de-

pending upon the number of samples and their radioactivity. Basically the material should be adequately shielded but should easily be identifiable and retrievable without undue exposure.

Pigs (Fig. 17-9) for individual sources are highly desirable. In some cases, a number of samples can be stored behind a lead-brick barricade. Labeled tags external to the shield but connected by string to the bottles behind the shield provide ready identification and the means of lifting the bottle from the barrier.

Built-in **storage wells** are used when many samples are to be stored. Individual pipes 4 to 8 in. in diameter are set horizontally or vertically in concrete. Samples are placed in individual pigs which are inserted in the pipes. Such a system provides storage and recovery with a minimum of dosage exposure.

Underwater storage is an economical technique for sealed or insoluble solid sources. Canals such as previously described are often used for storage. In some cases individual pipes set vertically in the ground or concrete are filled with water to provide shielding.

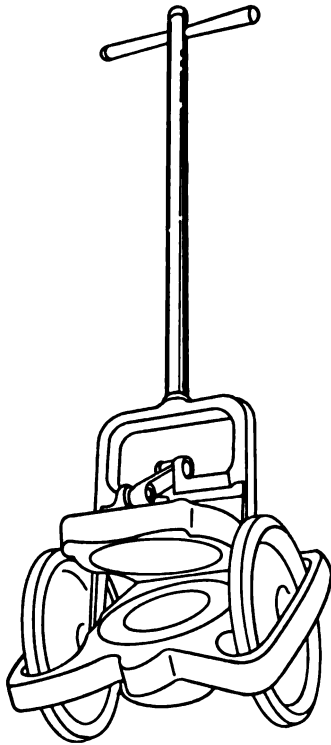


FIG. 17-37. Lead-shield cart for transporting applicators. (*Radium Chemical Company, Inc.*)

TRANSPORT AND SHIPMENT

The movement of radioactive material from one location to another involves both radiation-dosage control and contamination control.

Intramural transfer is usually accomplished with the use of a local shield or pig. Because of the weight, some sort of dolly is often required. A wheeled carrier (Fig.

17-37) designed for radium applicators facilitates transport. A movement of the handle opens the lid.

When many transfers are anticipated, special equipment must be built. At Oak Ridge National Laboratory, a shield mounted permanently on a lift truck is used to transfer trays of hot samples between hot cells. The lead shield or safe has a hydraulically operated sliding lift door. The shield is positioned against the cell door, and both doors open for the transfer.

The shipment of radioactive material on public carriers must meet certain standards of the Interstate Commerce Commission regarding radiation dosage rate, strength of container, ability to resist fire, etc. Oak Ridge National Laboratory has developed a series of radioisotope containers for shipment that economically meet these requirements.

Disposable shipping containers are basically 350-lb test corrugated fiber-board cartons with an insert to restrain a tin can containing the actual container of radio-

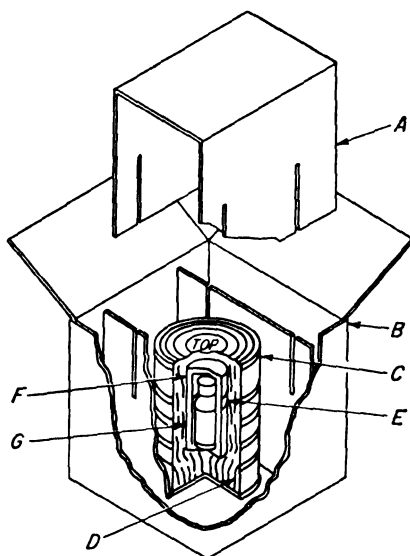


FIG. 17-38. Disposable shipping container for separated-liquid gamma emitters. (A) Fiberboard insert. (B) Fiberboard box. (C) Tin can. (D) Absorbent-paper wadding. (E) Masking-tape seal. (F) Top section of lead container. (G) Bottom section of lead container. (Selected Reference Material; *Chemical Processing and Equipment*, AEC Rept. TID-5276, p. 249.)

active material packed in absorbent-paper wadding. These containers are used for separated liquid beta emitters (e.g., up to 1 curie of P-32) and small amounts of gamma emitters (e.g., up to 30 mc of I-131) (Fig. 17-38). The use of disposable lead shields up to $\frac{1}{2}$ in. thick around the inner container allows the shipment of up to 600 mc of I-131.

Returnable shipping containers are used for gamma-emitting separated isotopes and for unseparated isotopes (Fig. 17-39). The separated isotopes container, de-

17-34 EQUIPMENT FOR HANDLING, STORAGE, AND TRANSPORTATION

pending upon the thickness of lead and the size of the box, can be used for up to 500 mc of Co-60. This container employs 3 in. lead shielding in a reinforced plywood box about 20 by 20 by 17 in. The other containers are made in thicknesses up to 7½ in. of lead which weigh about 1,300 lb. Eighteen curies of Co-60 can be shipped in this container.

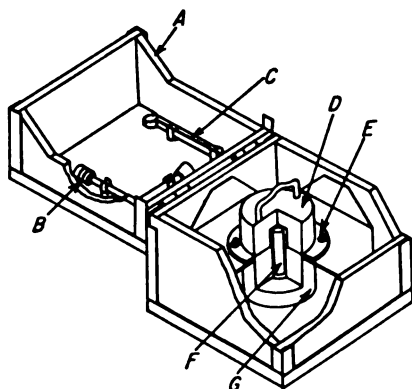


FIG. 17-39. Returnable shipping container for unseparated-solid beta and gamma emitters. (A) 3/4-in. plywood box. (B) Wrench. (C) Decapper. (D) Lead shield. (E) Nut. (F) Aluminum can. (G) Lead shield. (*Selected Reference Material; Chemical Processing and Equipment, AEC Rept. TID-5276, p. 251.*)

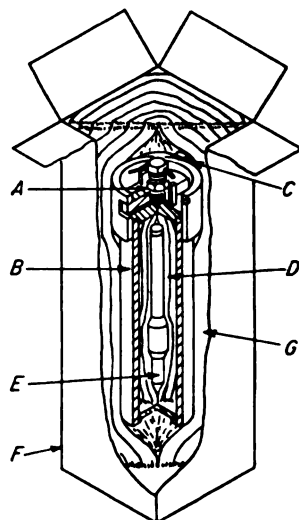


FIG. 17-40. Shipping container for gases. (A) Locking bolt. (B) Aluminum container. (C) Wire and seal. (D) Paper wadding. (E) Pyrex ampoule. (F) Fiberboard box. (G) Paper wadding. (*Selected Reference Material; Chemical Processing and Equipment, AEC Rept. TID-5276, p. 252.*)

Returnable gas containers (Fig. 17-40) are used for curie quantities of H-3 and millicurie quantities of A-37.

SPECIAL SHIPPING EQUIPMENT

The transportation of radioactive materials other than those of the isotope distribution system has required the development of special equipment and techniques.

Boxcars for bulk materials are in some cases modified by the installation of blowers in each end of the car. As the car is opened, the fans are activated to eliminate any radon that might have collected. Actual packaging of the bulk material is by steel or fiber drums.

Aircraft-wing-tip carriers are used by the British for unshielded shipments. Special compartments are constructed for special cylindrical containers about 1.7 in. in diameter by 5.2 in. in length. The maximum shipment is 300 mc.

General References

- Stephenson, R., "Introduction to Nuclear Engineering," McGraw-Hill, New York, 1954.
- Shamos, M. H., and S. G. Roth, "Industrial and Safety Problems of Nuclear Technology," Harper, New York, 1950.
- Bradford, J. R., "Radioisotopes in Industry," Reinhold, New York, 1953.
- "Safe Handling of Radioactive Isotopes," NBS Handbook 42, 1949.
- "Control and Removal of Radioactive Contamination in Laboratories," NBS Handbook 48, December, 1951.
- "Protection against Radiations from Radium, Cobalt-60 and Cesium-137," NBS Handbook 54, September, 1954.
- Evans, R. D., "Problems Associated with the Transportation of Radioactive Substances," National Research Council, Nuclear Science Series, Preliminary Report 11, 1951.
- "The Protection of Workers against Ionising Radiation," International Labor Office, Geneva, 1955.
- Parker, H. M., Health-physics, Instrumentation and Radiation Protection, *AEC Rept. MDDC-783*, March, 1947.
- McCallan, K. L., Preparation and Use of Neutron Sources, *Nucleonics*, vol. 5, no. 7, p. 11, 1949.
- Ward, D. R., Design of Laboratories for Safe Use of Radioisotopes, *AEC Rept. AECU-2226*, November, 1952.
- Selected Reference Material; Chemical Processing and Equipment, *AEC Rept. TID-5276*.
- Fourth Annual Symposium on Hot Laboratories and Equipment, held in Washington, D.C., Sept. 29 and 30, 1955, *AEC Rept. TID-5280*, 1955.
- Fifth Hot Laboratories and Equipment Conference, 1957 Nuclear Congress, Philadelphia, March, 1957.
- Tompkins, P. C., A. Broido, and J. D. Teresi, The Handling of Radioactive Materials in the Experimental Biology Section, *AEC Rept. MDDC-377*, September, 1946.
- Bizzell, O. M., Equipment for Radioisotope Laboratories, *AEC Rept. AECU-2875*, July, 1954.
- Apperly, A., Remote Handling Equipment, *Atomic Energy Research Establ. G. B. Rept. AERE E/R 1291*, 1953.
- Farmakes, J. R., Snare Type Handling Device, *Nucleonics*, vol. 10, no. 11, p. 90, November, 1952.
- Brown, G. E., Cobalt-60 Range for the Calibration of Dose and Dose Rate Meters, *Natl. Bur. Standards U.S. Rept. 2091*, December, 1952.
- Tompkins, E. R., Equipment for Radiochemical Operations, *AEC Isotopes Div. Circ. A-6*, June, 1948.
- Hot Labs—A Special Report, *Nucleonics*, vol. 12, no. 11, pp. 35-100, November, 1954.
- Goertz, R. C., and F. Bevilacqua, Remote Handling, in "Reactor Handbook: Engineering," chap. 7.1, pp. 855-873; *AEC Rept. AECD-3646*, May, 1955.
- Yecker, D. F., Hot Laboratories, in "Reactor Handbook: vol. 2, Engineering," chap. 7.3, pp. 909-923; *AEC Rept. AECD-3646*, May, 1955.
- Monk, G. S., K. R. Ferguson, and D. F. Yecker, Remote Viewing, in "Reactor Handbook: Engineering," chap. 7.2, pp. 785-908; *AEC Rept. AECD-3646*, May, 1955.
- Rupp, A. F., Methods of Handling Multikilocurie Quantities of Radioactive Materials, *Proc. Intern. Conf. Peaceful Uses Atomic Energy, Geneva*, vol. 14, pp. 128-135, 1956.

17-36 EQUIPMENT FOR HANDLING, STORAGE, AND TRANSPORTATION

Narver, David L., Jr., "High Unit Weight Concretes for Radiation Shielding," Holmes and Narver, Inc., Los Angeles.

Davis, H. S., How to Choose and Place Mixes for High-density Concrete Reactor Shields, *Nucleonics*, vol. 13, no. 6, pp. 60-65, June, 1955.

Burnett, J. R., R. C. Goertz, and W. M. Thompson, Mechanical Arms Incorporating a Sense of Feel for Conducting Experiments with Radioactive Materials, *Proc. Intern. Conf. Peaceful Uses Atomic Energy, Geneva*, vol. 14, pp. 116-121, 1956.

Tolbert, B. M., N. Garden, and P. T. Adams, Special Equipment for C¹⁴ Work, *Nucleonics*, vol. 11, no. 3, p. 56, March, 1953.

Morgan, G. W., Facilities and Equipment for Isotopes Program, *Hospitals*, March, 1955.

Guide for Selection of Equipment for Radioactivity Laboratories, *Nucleonics*, vol. 7, no. 5, pp. R-1-R-24, November, 1950.

Blatz, H., Transportation of Large Quantities of Radioactive Materials, *Proc. Intern. Conf. Peaceful Uses Atomic Energy, Geneva*, vol. 14, pp. 136-139, 1956.

A Manual of Remote Viewing, *AEC Rept. ANL-4903*.

Ferguson, K. R., Design and Construction of Shielding Windows, *Nucleonics*, vol. 10, no. 11, pp. 46-51, November, 1952.

Goertz, R. C., Fundamentals of General Purpose Remote Manipulators, *Nucleonics*, vol. 10, no. 11, pp. 36-42, November, 1952.

Ter-Pogossian, M., and A. I. Sherman, Handling of Radioactive Gold for Therapeutic Purposes, *Nucleonics*, vol. 10, no. 3, pp. 23-27.

Industrial Technical Information Meeting on Cold Processing of Enriched Uranium, Oak Ridge, Sept. 13-15, 1956, *AEC Rept. TID-7518* (pt. 1), September, 1956.

Kelman, L. R., W. D. Wilkinson, A. B. Shuck, and R. C. Goertz, Handling Alpha-active Pyrophoric Materials, pt. 1, *Nucleonics*, vol. 14, no. 3, pp. 61-65, March, 1956; pt. 2, *Nucleonics*, vol. 14, no. 4, pp. 65-71, April, 1956; pt. 3, *Nucleonics*, vol. 14, no. 5, pp. 77-82, May, 1956.

Portable Radiation Shielding Materials, *Nucleonics*, vol. 13, no. 5, pp. 84-86, May, 1955.

Lumbert, A. R., Low-radiation-level Metallurgy Cell, *Nucleonics*, vol. 12, no. 11, pp. 38-39, November, 1954.

Glen, H. M., Materials of Biological Shielding, 2d Nuclear Engineering and Science Conference, Paper 57-NESC-51, ASME, New York, 1957.

Spence, R., An Atomic Energy Radiochemical Laboratory—Design and Operating Experience, *Proc. Intern. Conf. Peaceful Uses Atomic Energy, Geneva*, vol. 7, pp. 39-43, 1956.

Fields, P. R., and C. H. Youngquist, Hot Laboratory Facilities for a Wide Variety of Radiochemical Problems, *Proc. Intern. Conf. Peaceful Uses Atomic Energy, Geneva*, vol. 7, pp. 44-48, 1956.

Pravdjuk, N. F., Metal-research "Hot Laboratory," *Proc. Intern. Conf. Peaceful Uses Atomic Energy, Geneva*, vol. 7, pp. 49-56, 1956.

Yakovlev, G. N., et al., A Hot Analytical Laboratory, *Proc. Intern. Conf. Peaceful Uses Atomic Energy, Geneva*, vol. 7, pp. 57-61, 1956.

Garden, N. B., Laboratory Handling of Radioactive Material, *Proc. Intern. Conf. Peaceful Uses Atomic Energy, Geneva*, vol. 7, pp. 62-66, 1956.

Dismuke, S. E., et al., Hot Laboratory Facilities and Techniques for Handling Radioactive Materials, *Proc. Intern. Conf. Peaceful Uses Atomic Energy, Geneva*, vol. 7, pp. 67-81, 1956.

Section 18

SURFACE CONTAMINATION AND DECONTAMINATION

By

PAUL C. TOMPKINS, Ph.D., *U.S. Naval Radiological Defense Laboratory, San Francisco, California.*

Introduction.....	18-2
Scope.....	18-2
Nature of Contamination.....	18-2
Theory and Test Methods.....	18-3
Contaminability and Decontaminability.....	18-3
Measurement of Contaminability.....	18-5
Measurement of Decontaminability.....	18-5
Selection of Materials.....	18-9
General Requirements.....	18-9
Walls.....	18-10
Floors.....	18-11
Benches.....	18-11
Handling Equipment and Shields.....	18-12
Clothing.....	18-12
Operations.....	18-12
Measurement and Evaluation of Surface Contami- nation.....	18-12
Decontamination.....	18-15
General Building Decontamination.....	18-19
Personnel Decontamination.....	18-19
Clothing.....	18-19
Minor Incidents (Spills).....	18-20
Major Incidents (Runaway Reactors).....	18-21
Special Definitions.....	18-22

SURFACE CONTAMINATION AND DECONTAMINATION

Paul C. Tompkins

References: "Nuclear Science Abstracts," "Biology and Medicine Category," "Radiation Hazards," and "Protection Subcategory," NBS Handbooks 47, 48, 49, 51, and 52. *Federal Register*, vol. 21, no. 6, Jan. 11, 1956.

INTRODUCTION

Scope

Radioactive contamination may be defined as the undesired presence of radioactive materials in amounts that may be considered harmful to the health and safety of personnel, or the validity of experiments or products. **Contamination** of a surface may be defined as the deposition and attachment of radioactive materials to surfaces. **Decontamination** of a surface may be defined as the removal of radioactive materials from the surface. The problems of surface contamination and decontamination will be considered primarily in their relationship to health and safety of personnel.

Nature of Contamination

Radioactive contamination, from its definition, results from the loss of amounts of material which are often inconsequential under normal conditions, but which have become important because they contain enough radioactivity to be detrimental. The nuclear properties (alpha, beta, gamma radiation, half-life, etc.) of the radionuclides are quite independent of the chemical properties of the material. Since the contaminability and/or decontaminability of a surface is directly related to the physical and chemical behavior of both the contaminant and the surface, these phenomena can be related to radioactivity measurements only under carefully prescribed conditions.

Sources of Contamination. The main sources of contamination leading to either contact (skin, hands) contamination, or loose (adsorption on dust and dirt) contamination are summarized in Table 18-1.

Some general relationships between radioactivity levels, the type of work being conducted, and the contamination precautions are summarized in Table 18-2.

Table 18-1. Sources of Radioactive Contamination

-
1. Chemical reactions leading to the loss of a gas
 2. Evaporation of a liquid
 3. Aeration of a liquid
 4. Liquid transfers of all kinds
 5. Grinding or manipulation of a solid
 6. Recoil from radioactive disintegration (especially with α emitters)
 7. Adsorption on surfaces (especially high-specific-activity material)
-

THEORY AND TEST METHODS

Contaminability and Decontaminability

Contaminability of a surface is its tendency to hold radioactive materials with which it has come in contact. In practice, it makes little difference whether this tendency to hold materials is due to purely physical causes (e.g., trapping in a crack) or comes from chemical causes (e.g., adsorption or ion exchange). Factors which contribute to a high surface contaminability are:

1. Porosity
2. Roughness
3. Wettability
4. Specific chemical reactions between a radioelement and the surface material
5. Adsorption of counter ions which establish an electrical double layer favorable to the attachment of the chemical form of the radioelement present

As met in industrial and laboratory practice, all loose material is first removed, usually with water or by vacuum cleaning, and the **relative contaminability** is measured in terms of the fraction of the original deposit which then remains. Thus, **surface contaminability** is a function of its **decontaminability**, i.e., the fraction of material initially deposited which is not removed.

In the case of **liquid-drop contamination**, when the liquid first contacts the surface, it immediately spreads over an area, the extent of which depends on the surface properties of both the liquid and the surface. Ionic and colloidal components immediately begin to react with or adsorb on the surface very rapidly at first but then more slowly as the layer at the interface becomes depleted. Also, capillary action pulls the liquid into both macro- and micropores and indentations. Evaporation sets in, particularly at the edges of the drop. The combined adsorption-diffusion process continues until the residual solution becomes saturated. After this stage, the remaining material simply crystallizes and deposits on top of the surface.

When the surface is subsequently treated with a **liquid cleaning agent**, the same sequence of liquid-interface reactions is initiated. The fraction of the total deposit which can be drawn into the bulk volume of the cleaning agent thus depends on the ability of that agent to dissolve and/or displace adsorbed material from the surface.

Table 18-2. Effect of Radioactivity Level on Contamination *

Level.....	<1 μ c	1 μ c to 1 mc	1 mc to 1 curie	>1 curie
Features of technique...	Quantitative technique 99.9% control, aseptic precautions	99.999% control segregation of equipment to avoid cross contamination, avoid spattering and solution losses	99.999999% control, enclosed operations, either equipment or area	Extreme control, complete enclosure, buffer areas
Special features of laboratory and equipment, in addition to aseptic precautions	Ordinary laboratory	Radiochemical hood desirable, storage and cleaning facilities duplicated, protect bench tops	Radiochemical hood mandatory, dry boxes advisable. Cleanable ducts, special waste disposal. High-efficiency hood and dry-box filters	Decontaminable ducts and drains, dry box or isolated cubical with high-efficiency filters
Special problems.....	Protection of experiments in vicinity of higher-level operations	Adsorption of high specific activity material; hand contamination and second-order transfer	Decontaminatable surfaces; accumulation (long-lived activity); ventilation, protection against accidents important	Gas, liquid, and solid waste storage or disposal
Examples.....	Tracer-level experiments with β -counting	Tracer-level experiments with γ -activity counting; preparation of stocks of β -tracer, process tracer application. Isolation of a radionuclide which is a minor constituent of a mixture	Preparation of stocks of γ -tracer, high-level C^{14} synthesis, therapeutic work. Isolation of a radionuclide which is a minor constituent of a mixture	Radioactive sources for radiography, etc.

* Tompkins, P. C., and H. A. Levy, Impact of Radioactivity on Chemical Laboratory Techniques and Design, *Ind. Eng. Chem.*, vol. 41, p. 228, 1949.

Measurement of Contaminability

Contaminability of a material has usually been measured by one of several empirical methods developed at various laboratories. Following are examples of several of these methods which have been published.

Isotope-adsorption Method. (Tompkins, P. C., and O. M. Bizzell, Working Surfaces for Radiochemical Laboratories, *Ind. Eng. Chem.*, vol. 42, p. 1469, 1950; vol. 42, p. 1475, 1950.) Pipette 0.1 ml of the contaminant onto a sample and cover to prevent evaporation. Allow to stand for 1 hr. Carefully withdraw the drop with a transfer pipette and wash by 10-sec immersions in four successive beakers of water, followed by 10-min immersion in a fifth. Dry over CaSO_4 and determine the residual activity.

$$\text{Calculation: \% adsorbed} = \frac{\text{activity remaining}}{\text{activity added}} \times 100$$

Isotope-adsorption Method. (Parker, G. W., and G. M. Herbert, Decontamination Characteristics of Porcelain Enamel, *Nucleonics*, vol. 12, no. 11, p. 72, 1954.) Pipette 0.1 ml of the contaminant onto a sample and dry about 6 hr without heating. Flush 1 min with 2 to 3 gal/min of water flowing across the sample. Dry and determine the residual activity. The calculation is the same as for the preceding procedure.

Liquid-hold-up Method. (Gevantman, L. H., B. Singer, T. H. Shirasawa, and H. K. Chan, "Liquid Hold-up As a Measure of Contaminability," to be published.) The test sample is weighed, immersed in a beaker of water containing contaminant for a graded length of time, drained a few seconds, and reweighed. The liquid retention is calculated in terms of $\mu\text{l}/\text{cm}^2$ of sample. If the weight difference is low, the sample is sealed in a closed container with a weighed amount of desiccant in excess of that required to combine with water on sample. The weight increase of the desiccant is then measured.

Table 18-3 illustrates the general contaminability ranges of common materials. The data have been selected from a variety of sources which are not completely comparable. The information listed represents the author's personal judgment as to the most likely result. It must be recognized that, under some specific conditions, quite different results may be observed. Therefore, it is usually desirable for each organization to measure the **contaminability of materials** under the conditions and with the radioelements which will actually be employed whenever these facts can be determined.

Measurement of Decontaminability

Decontaminability is a measure of the extent to which deposited radionuclides can be removed from a surface. For **nonporous materials** being treated with liquids, the effectiveness of a method will depend on the **partition** of the contaminant between the surface and the liquid.

One would expect that the first action of a **liquid reagent** would be simply to dissolve the salts that remain unattached to the surface. Simultaneously, there would be reactions between constituents of the cleaning solution and radionuclides ad-

Table 18-3. Contaminability to Fission Products

	<10%	10-30%	30-70%	>70%
Polythene, Teflon, and Saran	All normal conditions			
Water-repellent paints (unweathered)		When oxidized or covered with foreign deposits		
Formaldehyde resins		Most solutions		
Glass (plate).....		Aged fission products		
Lead	With Ba^{140} in HCl	Surface scarred		With $\text{H}_2\text{P}^{32}\text{O}_4^-$, aged fission products
Polished stainless steel (type 147)	Dilute acid solutions of carrier-free P^{32} , Ba^{140} , and I^{131}		Aged fission products in dilute acid	
Unpolished stainless steel	Unblemished	Blemished		
Structural steel		Dilute acid solutions	With P^{32}	
Glazed-tile surfaces		When coated with laboratory or industrial film deposits		
Best grades of floor tile			Floor tiles in good condition	Tiles scarred or scratched (worn)
Plastic spray coats	Dried without pinholes	Pinholes on evaporation of solvent		
Alberene stone....		Impregnated with vinyl resins	Impregnated with formaldehyde or phenolic resins	Untreated
Transite.....				Untreated, contaminated with most acid solutions
Concrete	Painted, unworn	Painted, worn	Smooth but not impregnated. Exposed to insoluble hydroxides under alkaline conditions	Untreated, contaminated with most acid solutions

sorbed on the surface. Ion-exchange reactions and the formation of un-ionized complexes are both important types of reactions in this situation. The effectiveness will depend on both the equilibrium distribution and the reaction rates.

The decontamination process is more complicated with **porous materials**. Here, the contaminant must not only be desorbed or displaced by constituents of the cleaning solution, but it must also be removed from the pores. This involves **diffusion** of the solubilized contaminant, which is a relatively slow process as compared with displacement from non-porous surfaces.

Liquid-decontamination methods, therefore, lead to extremely complex physico-chemical systems involving the radioelement-surface-solution interactions. Such systems often show a high degree of specificity. The most successful practical methods exploit those particular mechanisms within this system which lead to a minimum percentage of the radioelement appearing as "tenacious" contamination.

Three general cases are usually encountered.

1. The radioelement is in a small volume of liquid and dries after it reaches the surface.

2. The radioelement is in a larger volume of liquid and is wiped off before it dries. The remainder then dries before it is cleaned.

3. The radioelement is air-borne and has become attached to dust or other particulates in the air before depositing on the surface.

The apparent cleaning efficiency increases in the same order. For example, **vacuum cleaning** is usually quite effective when dealing with the third general case.

Liquid cleaning procedures may involve covering the surface with the reagent, which is then allowed to stand a short time before it is drawn off, or the solution may be allowed to run over the surface continuously. In either case mechanical action such as **stirring** may be used.

Two typical results of a **stepwise decontamination sequence** are shown in Fig. 18-1. The solid line is typical of what is observed when a smooth impervious surface

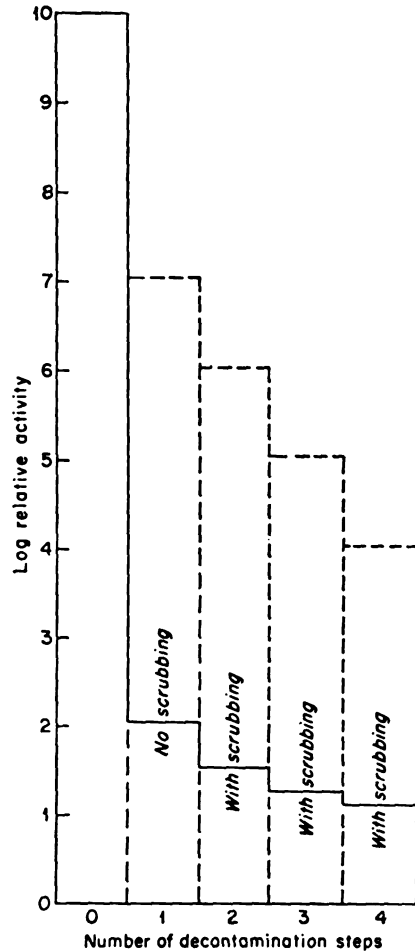


FIG. 18-1. Relative activity remaining on surface after stepwise decontamination sequence.

(e.g., glass) is cleaned with acid or a complexing agent. Most of the material which can be removed is taken off almost immediately with the first application. This is generally interpreted as representing the amount in the superficial unattached layer, and almost immediately establishes the magnitude of the final result. If the requirement is a reduction of radiation intensity to a relative level of 10 (log relative activity = 1, in Fig. 18-1), and the first treatment reduces the level to a relative intensity of 100 (log relative activity = 2), repeated treatment with scrubbing may be successful.

A second common observation, particularly when metals are treated with acids, is that the first step reduces the radioactivity level to that shown by the dotted line. The application of the second step reduces the radioactivity to that represented by the solid line. These results appear to be associated with the removal of the **superficial oxide layer**. If the surface activity at this time is not within a factor of 5 or 10 of the final level required, the method should be changed because further treatment with the same reagent will only result in more erosion to the surface without a comparable removal of the radioelement.

A third common observation follows the sequence shown by the dotted lines. This pattern of fairly large successive reductions in the surface activity (a factor of 10) is most often observed when **painted surfaces** are cleaned with a mild reagent and the contaminant itself is noncorrosive.

Decontamination test methods are usually designed to compare the relative effectiveness of various methods and conditions for a number of contaminants and surfaces. The variables studied include the element, form, and composition of the contaminating medium, method of application, and the process of cleaning as performed on a laboratory scale. Such tests give a measure of the maximum decontamination that can be expected under ideal conditions.

In actual practice where large areas are involved, additional factors are encountered. After the contaminant has been detached from the surface, it must be flushed off and disposed of. This step usually limits the effectiveness of methods to values well below those predicted from the laboratory scale experiments.

Results are usually reported in one of the following ways.

Decontamination Factor. (Parker, G. W., and G. M. Herbert, Decontamination Characteristics of Porcelain Enamel, *Nucleonics*, vol. 12, no. 11, p. 72, 1954.)

$$\text{Decontamination factor} = \frac{\text{initial activity}}{\text{final activity}}$$

Decontamination Index. (Tompkins, P. C., O. M. Bizzell, and C. D. Watson, Practical Aspects of Surface Decontamination, *Nucleonics*, vol. 7, no. 2, p. 42, 1950.)

$$\text{Decontamination index} = \log \frac{\text{initial activity}}{\text{final activity after 2 steps of decontamination}}$$

Spill Index. (Tompkins, P. C., O. M. Bizzell, and C. D. Watson, Practical Aspects of Surface Decontamination, *Nucleonics*, vol. 7, no. 2, p. 42, 1950.)

$$\text{Spill index} = -(\log \% \text{ adsorbed} - \text{decontamination index})$$

One must use care in examining the literature of this subject since most studies are made with a certain application in mind. Therefore, the experimental conditions

for which numerical values are reported vary widely and are seldom directly comparable.

Typical procedure: Pipette 0.1 ml of solution onto the sample of material and dry by evaporation in air. Add reagent, allow to stand 3 min, remove, and wash the sample with water. Dry and count. Repeat the procedure with 2 ml of reagent and brush the surface manually with a surgical scrub brush. Flush with water, dry, and determine the activity. This procedure has been used in the determination of the **decontamination index**.

SELECTION OF MATERIALS

General Requirements

Normally, there are four general requirements for materials used in facilities and equipment. These requirements arise from considerations of shock and abrasion, weathering, fabrication, and appearance. The nucleonics industry is faced with four additional requirements arising from considerations of susceptibility to contamination, decontaminability, resistance to corrosion and heat, and shielding. (Terrill, J. G., "Surfaces and Finishes for Radioactive Laboratories," Research Correlation Conference No. 3, National Academy of Sciences, Washington, D.C., November, 1951).

The selection of materials depends to a large degree on the construction, maintenance, and **contamination-control** policies that are adopted. Wherever possible, the best policy is to protect surfaces against contamination, and prevent its penetration. For example, protective coatings for porous materials are built up as follows:

First Layer—Seal Coat. This coating must penetrate porous materials to some depth (several millimeters on concrete or wood) and fill all pores with a homogeneous, chemically resistant body.

Second Layer—Permanent Surface Coat. Preferably a homogeneous extension of the seal coat. It should be resistant to the conditions for which it will be used and should have a low susceptibility to the radioelements of interest.

Third Layer—Final Surface Coat. This coating may be varied to suit conditions. In addition to the necessary physical and chemical properties it should have a low susceptibility to contamination.

Generally, **physical considerations** predominate over chemical considerations in practice. Unless cracks, holes, and sharp corners are eliminated, and the surfaces are free of pinholes, chips, and scratches, material tends to get drawn into them and become inaccessible to cleaning methods.

In many applications, it is better to use cheap, easily replaced materials than to use a fixed installation which must be cleaned. This is particularly true for floors and other areas that are subjected to abrasion and wear. A good smooth physical surface (usually painted) is the main requirement. Ordinary industrial and household paint finishes are usually quite porous when aged and are inferior to rubber- or vinyl-base types. Because of the deposition of smoke, soot, and dirt, a layer soon forms on which contaminants settle. As long as this film can be washed off, the chemical composition of the underlying surface (paint, glazed tile, etc.) is relatively

18-10 SURFACE CONTAMINATION AND DECONTAMINATION

unimportant. In general, smooth nonporous surfaces can be expected to decontaminate with the same efficiency after a number of decontaminations, whereas rough, porous, or corrosible surfaces will decontaminate with decreasing efficiency.

All areas subjected to high contamination are usually covered with a disposable exterior. All metals and wood should be given a **protective coating**, the last layer of which is a film of highly water repellent material such as **wax** or one of the **silicone** compounds.

Table 18-4. Approximate Spill Indexes (S.I.) and Decontamination Indexes (D.I.) for Different Levels of Activity *

Approx activity level	S.I.	D.I.
Less than 1 μ c.....	~3	2
1 μ c-1 mc (low frequency).....	5-6	3-4
1 μ c-1 mc (high frequency).....	7-8	5-6
1-100 mc (low frequency).....	7-8	5-6
1-100 mc (high frequency).....	9-10	7-8
100 mc-1 curie (low frequency).....	9-10	7-8
100 mc-1 curie (high frequency).....	11-12	9-10

* Taken from *Ind. Eng. Chem.*, vol. 42, no. 8, July 29, 1950.

Table 18-4 shows the desirable **spill indexes**, and the **decontamination indexes** which materials should have in relation to the activity levels. Table 18-3 offers a general guide as to the selection of material. However, one should turn to the original literature for data on specific materials. Direct evaluation in one's own laboratory is the most reliable basis on which to select the combination of materials most suited to a particular application.

Walls

Surfaces for walls and partitions can be readily divided into coatings which are applied after installation and prefinished partitions which have a factory-applied coating. Partition material is preferably of wood, metal, or blocks of concrete. Plaster and tile generate too much dust to be generally acceptable. A typical wall surface might consist of wood covered by two coats of a suitable vinyl-base paint (Prufcoat, Amercoat, etc.) followed by a stripper base over which a **strip coat** could be applied if needed. Cracks would be covered with polyethylene tape prior to painting. Masonite, transite, plywood, and metal sheeting are amenable to this arrangement. Metal partitions with baked-enamel surfaces are generally readily cleaned if the cracks are properly sealed with polyethylene tape or the equivalent. Protection of such materials is desirable even when the maintenance policy is one of replacement, since small amounts of the contaminant can be removed and the probability of gross contamination is low and since replacement is generally inconvenient. General maintenance consists of frequent dusting or vacuum cleaning plus washing as necessary with a good detergent and warm water. The surfaces should be kept in good condition at all times.

Floors

Experience with continuous floorings such as concrete surfaced with troweled mixtures of mineral aggregate bonded with asphaltic, synthetic rubber, or resinous compounds has been generally unsatisfactory. Therefore, these surfaces are used only where heavy loading and abrasion resistance supersede the contamination criteria. Semicontinuous flooring such as linoleum and rubber sheeting requires considerable maintenance. These materials usually suffer from inadequate resistance to the chemicals and solvents commonly used with radiochemical-processing methods. The preferred flooring is usually one or another of the various tiles which are available; although normal decontaminability is usually very poor, replacement of contaminated sections is relatively easy and inexpensive. The performance ratings from the contamination point of view may be considered to decrease in the general order: vinyl tile > rubber tile > asphalt tile > ceramic tile > linoleum and other sheetings > continuous concrete. Frequent waxing and cleaning are the basic general maintenance requirement.

Benches

The early emphasis on stainless steel has been replaced by emphasis on plastic-coated surfaces since the stainless steel has proved to be susceptible to pitting by corrosion with a corresponding decrease in decontaminability. The composition of the base material (e.g., stone, wood, masonite, transite) is relatively incidental from the contamination viewpoint if the coating forms a continuous covering. Table 18-5 shows some of the materials which have been recommended and the use to which they can be put. Working surfaces are normally protected by a disposable

Table 18-5. Suggested List of Materials and Area of Application *

<i>Material class</i>	<i>Suggested use</i>
Polythene sheet.....	Hot sinks, bench tops, trays, hood interiors, drain lines, ductwork and process lines, and vessels where contamination is a marked problem
Air-dried vinyl coatings....	General use on floors, walls, ceilings, and equipment
Baked enamel and other heat-treated surfaces	Laboratory furniture and removable partitions; metal equipment of all kinds where pressure and abrasion are not encountered
Floor tiles.....	Sanitile was best of group tested. Better ones desired
Alberene coated with vinyl, VMCH, Saran sheet	Bench tops

* Taken from Watson, C. D., T. H. Handley, and G. A. West, Decontamination and Corrosion Resistance Properties of Selected Laboratory Surfaces, *Oak Ridge Natl. Lab. Rept. 732*, August, 1950.

18-12 SURFACE CONTAMINATION AND DECONTAMINATION

layer such as paper. With proper care, such surfaces should rarely become severely contaminated except in the high-level enclosures.

Handling Equipment and Shields

Such equipment is usually of metal, the surfaces of which are untreated when used as bearing surfaces. As a minimum, all such objects should be pickled with an appropriate corrosion-resistant agent, and the exterior treated with a water-repellent material such as a silicone or wax. Such treatment is particularly important for parts of structural steel, brass, aluminum, and magnesium which will otherwise pit rapidly. A strip coat is often advisable.

Clothing

Experience with clothing is quite variable. However, treatment of clothing for water-repellent agents has not been explored to the point where such treatment is a standard recommendation. Clothing is usually considered expendable and is replaced when normal cleaning plus radioactive decay is inadequate.

OPERATIONS

Measurement and Evaluation of Surface Contamination

Measurements of surface activity are normally taken for purposes of **radiation exposure** and **contamination control**. For the former purpose a direct reading with an instrument is made; for the latter, a wipe test is usually employed. The instruments and their general calibration are described in Sec. 9. Under normal working conditions, alpha emitters and most beta emitters may be disregarded in determining danger from radiation exposure. The monitoring instrument should respond only to the gamma field. For locating beta or alpha emitters, instruments sensitive to these radiations are useful. **Several critical points must be borne in mind in using these instruments for measuring surface contamination.** These are as follows:

1. The instrument measures only those radiations which enter the detection unit.
2. The interpretations are desired in terms of activity on the surface, and the calibration methods are not a faithful reproduction of the radionuclide form or distribution being measured.
3. With alpha-emitting isotopes, allowance must be made for the short range of the alpha particles and the possibility that only a small fraction of the radiation being emitted is actually measured by the detector.
4. With beta-emitting isotopes, the reading-to-true-surface-activity relationship is sensitive to the beta energy.
5. With gamma-emitting isotopes, the radiations may originate from a location other than from the immediate surface being measured.
6. The source-to-detector geometry is quite variable because contamination is usually spotty and inhomogeneous in its distribution on the surface, and the effective detector volume never corresponds exactly to its physical boundaries.

Assuming a reasonably uniform distribution of activity, three major factors influence a given meter reading, particularly when measuring beta radiations. These are:

1. Area of contamination
2. Energy spectra of the radiations
3. Distance between the surface and the instrument

Table 18-6. Ratio $\frac{\text{rads/hr to skin}}{\text{r/hr on the meter}}$ *

G-M meters				Ion chambers			
Instru- ment †	Area	Tl ²⁰⁴ (0.78 Mev)	P ³² (1.7 Mev)	Instru- ment ‡	Area	Tl ²⁰⁴ (0.78 Mev)	P ³² (1.7 Mev)
A	6 by 6 in.	18	5	A	4½ by 6 in.	9	9
	1 in. diam	200	63		1 in. diam	125	125
B	6 by 6 in.	125	4.5	B	4½ by 6 in.	12.5	8.3
	1 in. diam	200	71		1 in. diam	200	125
C	6 by 6 in.	13	5	C	4½ by 6 in.	12.5	8.3
	1 in. diam	200	77		1 in. diam	330	200
D	6 by 6 in.	24	9				
	1 in. diam	200	110				
E	6 by 6 in.	48	8.3				
	1 in. diam	1,000	125				

* Taken from Krakauer, S., H. Byfield, F. Legler, and G. Failla, Review of Characteristics of Health Monitoring Instruments, Calibration of Survey Instruments by Plane Thin Beta Sources and Natural Uranium, *AEC Rept. NYO-5642*, Apr. 1, 1955; and Krakauer, S., "Calibration of Survey Instruments SII-17C, SIC-17D, and CD-V-700 in Terms of Dose Rate at Contact Using Plane Thin Sources (P³², Sr⁹⁰-Y⁹⁰)," Radiological Research Laboratory, Columbia University, 1955.

† Different instruments. Distance 1 in. from plaque to meter. Window thickness 30 mg/cm².

‡ Different instruments. Contact with the plaque. Window thickness 30 mg/cm².

Table 18-6 shows the effect of area and energy at a distance of 1 in. from a plaque to the detector for a beta-emitting nuclide. The effect would be even greater if the plaque area were smaller than 1 in. For the 6-in.-diameter source, the ratio is between 30 and 40 at 6 in. instead of between 5 and 10 as it is at 1 in. With the smaller-area source, the ratio would increase even more sharply. At best, the measurement of surface activity is only an approximation of the true activity present, and great care in the application of correction factors is required to make the interpretations reliable within an error of 100 per cent. **The following procedure is particularly applicable to beta monitoring** and if rigorously followed will do much to prevent major errors of interpretation.

1. Keep the shield open, and hold the detector at a distance of 1 in. from the surface being measured or as close as possible when measuring alphas.

18-14 SURFACE CONTAMINATION AND DECONTAMINATION

2. Pass the probe over every portion of the surface being examined, but avoid actually touching objects with any part of the instrument.

3. When measuring low levels, proceed slowly and carefully. Stop and hold the meter still long enough for the meter to reach its maximum deflection.

4. Be sure the exposed portion of G-M tube instruments, or the open window of other types of instruments, faces the source when the measurement is being taken.

5. Check both background and zero positioning periodically.

6. If a contamination level that causes the meter to go off scale is encountered, the use of a higher-range instrument is better than taking measurements at a greater distance and hoping to correct by inverse-square calculations.

When accurately known standards are available, and the energy of the isotope being measured is known, a direct-comparison method is preferred. The requirements for the standard are:

1. The isotope is homogeneously distributed.

2. The source dimensions are comparable with the dimensions of the window.

3. The source is calibrated by film or extrapolation-chamber methods in terms of reps/hr at contact with the sample.

The beta-monitoring procedure is to calibrate the instrument with a known gamma-emitting source and obtain an instrument reading close to the center of the scale. Then substitute the beta source for the gamma source and move to a position that gives the same reading on the instrument. Measure the distance from source to instrument. If surface readings are then taken at this distance, and knowing the mr/hr reading and the mreps/hr at the surface of the standard source which produces this reading, the $\frac{\text{mreps/hr at the surface}}{\text{mr/hr on the instrument}}$ can be used to evaluate the activity on the surface as long as the emission characteristics of the unknown are similar to those of the standard.

The instrument measures the **total contamination** on the surface. Several studies have been made to relate the total activity on the surface to air-borne activity levels, to personnel contamination, and to amounts ingested. However, no consistent relationship appears to exist; so that evaluating the biological hazard from these measurements is not considered reliable, except when estimating direct contact dose to the skin. (Eisenbud, Merrill, H. Blatz, and E. V. Barry, How Important Is Surface Contamination? *Nucleonics*, vol. 12, no. 8, p. 12, 1954.)

The portion that is important from the practical point of view is the **loose contamination**, since this is the material that can be easily rubbed off, and which will be carried to other locations. It is possible to detect removable material by wiping a clean (uncontaminated) material such as absorbent paper (paper hand towels or toilet tissue) or filter paper (e.g., Whatman No. 1) over the object in question. The material wiped off may then be measured by any standard procedure using either survey instruments or regular laboratory counting equipment.

No standard **wiping technique** has been established; so each radiological hygienist will have to calibrate his own. One should standardize on the wipe material used, the pressure, area wiped, and counting procedure. Light pressure with the fingertips, and an area of 12 in.² are employed by most people. Systematic investigations at the U.S.NRDL show that the amount wiped off is generally proportional to the initial activity on the surface, whether the surface was previously cleaned or not. However, the proportionality constant depends on both the type of contaminant

and the previous treatment. The fraction removed by the wipe seems to center in the range of 0.1 to ~10 per cent of the initial activity and duplicate samples may differ by as much as 200 per cent. Nevertheless, for the kinds of decisions that need to be made, the test is often informative; with alpha emitters at low levels, it is the most informative test available. However, one must be careful to infer only what is justified by the approximate nature of the test.

Gummed paper or sticky tape has been used as an alternative to the wipe or smear method. Evidence to date does not justify the conclusion that the results can be reliably compared with the smear method. There is some evidence that the method is less reproducible than the wipe method, particularly when applied to smooth surfaces.

Decontamination

Routine decontamination methods, no matter to what they are applied, should emphasize thorough cleaning of the surface without breaking its superficial in-

Table 18-7. General Reagents for Decontamination

Reagent type	Typical representative	Where used	Remarks
Mineral acids . .	HCl, HNO ₃ , H ₂ SO ₄	Metals, glass, plastics, paint	10% solutions for dilute reagents. Destructive to surface. Usually most efficient reagents if materials can tolerate their use
Strong oxidizing solutions	Chromic acid cleaning solution	Glass, metals	Laboratory cleaning agent. Last-resort treatment. Often used for Pu
Trisodium phosphate		Glass, paints, plastics	10% solution. Often used hot. Removes wax and is hard on paint
Fluoride.....	(NH ₄) ₂ SiF ₆	Glass, porcelain, and other silicone materials	0.1-1% solution
Complexing agents	Sodium citrate, versene	All materials	1-10% solutions. Efficiency variable and pH-dependent. Often preferred choice for general use
Detergents.....	Household or commercial cleaning detergents such as Tide	All materials	Efficiency variable. Choice sensitive to water properties as well as materials and contaminants used with scrubbing

Table 18-8. Preliminary Summary of Building-material Decontamination Procedures

Procedures, equipment, and additives	Techniques and surfaces	Advantages	Limitations
1. Vacuuming, using appropriate-capacity industrial-type vacuum cleaner plus suitable exhaust filter. No additives	Conventional procedure. Applies to almost any surface covered by dry loose contaminant	Retains residue and filters exhaust, reducing aerosols. Often the first step	Applicable only to dry loose contaminant. Face masks advisable. Slow rate, therefore requires increased manpower and equipment for large areas
2. Hosing, using hose and nozzle. No additives	Progress from low- to high-contamination areas taking advantage of available drainage. Applies to building and floor materials	Inexpensive procedure for gross decontamination. Minimum protective clothing	Consumes large volumes of water. Runoff must be controlled
3. Hosing—hand scrubbing, using hose and nozzle plus floor brushes and mop buckets. Additives: detergents and complexing agents	Hose, progressing from low- to high-contamination areas taking advantage of available drainage, then hand scrub with solution containing 1% additive, and follow with hose flush. Applies to building materials	Effective procedure on dirty and greasy surfaces. Additives easily stored, nontoxic, noncorrosive, minimum protective clothing	Rate slower than (2). More manpower than (2). Additives use up storage space and not always available
4. Hot-liquid-jet cleaning—hand scrubbing—hosing, using steam injector, lance, floor brushes and hose, boots and face shields. Additives: detergents and complexing agents	Educt solution (containing 20% additive) with injector and scrub simultaneously, then follow with hose flush. Applies to building and floor materials	Same as (3) except for reduction in runoff	Requires special equipment and steam source. Otherwise same as (3). Steam introduces personnel hazard. Moderate protective clothing required

5. Steam cleaning, using hose and steam lance, boots and face shields. Additives: detergents and complexing agents	Additives introduced into flow system to produce 1% solution at lance, proceed as in (2). Applies to building and floor materials; major equipment	Reduced runoff, low manpower, otherwise same as (3)	Requires steam source. Rate lower than (2). Additives use up storage space and not always available. Steam introduces personnel hazard. Moderate protective clothing required
6. Stripping—hosing, using drum pump or spray pot, air and liquid hose, lance, hose and nozzle, protective hoods with air lines, boots and rubber gloves. Additives: Sodium, calcium, or potassium hydroxide containing thickening agents such as starch	Spray 10 to 20% caustic solution containing thickening agent 24 hr in advance of hose flush, proceed as in (2). Applies to painted surfaces	Highly effective, removes paint down to base surface. Faster than sandblasting	Requires special equipment and maximum protective clothing. Hazardous to personnel. Storage and availability problems. Will not remove rust. Cannot be used on aluminum and magnesium surfaces
7. Paint stripping and hosing, using same equipment as (6) and (8). Additive: cresylic acid	Same as (6) and (8) except standing time not established. Applies to painted surfaces	Same as (6) and (8). Removes certain paints not affected by caustics. Can be used on aluminum and magnesium surfaces	Additive more expensive than caustic, otherwise same as (6)
8. Paint stripping—steam cleaning, using equipment same as (6) except steam hose and lance replace firehose. Additives same as (6)	Spray caustic 24 hr in advance of steam cleaning. Applies to painted surfaces	Reduced runoff. Otherwise same as (6)	Same as (6)
9. Degreasing—hand scrubbing—rinsing, using spray rig or hand applicators, scrub brushes and water hose or steam line. Additives: emulsifying type degreasers	Apply degreaser, scrub and rinse (before compound dries) with water detergent solution or steam. Applies to machinery, vehicles, etc.	Removes heavy grease, oil, and dirt deposits not affected by detergents	Rate slower than (2). More manpower than (2). Additives use up storage space and not always available

Table 18-8. Preliminary Summary of Building-material Decontamination Procedures
(Continued)

Procedures, equipment, and additives	Techniques and surfaces	Advantages	Limitations
10. Dipping—rinsing, using crane or hoist, dip tank and rinse tank, or hosing or steaming facility. Additives: Same as (6), (7), and (9)	Immerse in suitable additive until grease or paint has been removed and rinse. Applies to portable gear such as pumps, engines, etc.	May be handled remotely, otherwise same as (6), (7), (8), and (9). Solutions usually reusable over many applications	Requires heavy equipment and therefore increased manpower. Limits size of objects decontaminated
11. Rust removing—rinsing, using swabs, scrub brushes, and water hose or steam line. Additives: inhibited hydrochloric acid	Coat rust areas with remover, scrub and rinse. Applies to rusty surfaces	Replaces need for more expensive sandblasting in certain instances	Hazardous, slow, not so thorough as sandblasting. Requires maximum protective clothing
12. Sandblasting, using sandblasting machine, air and sand hose, blast hoods with air lines. Additives: Water may be introduced to reduce aerosols	A standard industrial procedure. Applies to building material such as metal, wood, stone, concrete, etc. Painted and/or rusty surfaces	Effective, removes rust as well as paint	Extremely slow, expensive, creates aerosol hazard. Maximum protective clothing
13. Vacuum blasting using vacuum blasting machine. Additives: special abrasive shot	Conventional procedure. Applies to building material such as metal, wood, stone, concrete, etc. Painted and/or rusty surfaces	Vacuum residue, reducing aerosols, otherwise same as (2). Recycles and reuses abrasive	Slow, awkward for areas other than horizontal. Face masks advisable. Auxiliary exhaust filters required
14. Floor refinishing, using floor-resurfacing machines. No additives	Planes off layer of material through action of rotary cutting tools. Applies to wood, concrete, brick, etc.	Only means short of demolition in some cases	Requires follow-up method to retrieve residue. Slow. Surface destructive. Aerosol hazard. Face masks required

tegrity. If noncorrosive methods prove inadequate, then somewhat harsher treatment may be used. However, care should be taken to employ a treatment that is as mild as possible and which will still be successful. Table 18-7 summarizes a list of typical reagents that have been used. These reagents are generally used with smooth impervious materials. It can be expected that the major effect is going to be produced within a few minutes, particularly if the area is scrubbed with a soft brush. Mild reagents and extensive scrubbing are generally successful in reducing the contamination to any desired level and are normally preferred to excessively harsh reagents. The amount of active material redeposited in adjacent areas is relatively minor if the operation is executed rapidly and carefully. Protective clothing and gloves should be worn during all such operations.

Miscellaneous Methods. Immersion methods are generally employed for laboratory equipment such as glassware, tongs, etc., and are more a problem of the radiochemical laboratory than of the radiological hygienist.

Ultrasonic units have been used for very special purposes, particularly for very small samples. (Feldman, M. J., *Advantages of Ultrasonic Cleaning*, *Nucleonics*, vol. 12, no. 11, p. 65, 1954.)

General Building Decontamination

General building decontamination usually emphasizes the surface-erosion methods and normally requires resurfacing of the affected area. Applicable methods and areas of use are given in Table 18-8. Wet methods are never recommended for use until vacuum cleaning and dusting with a wet cloth cease to be productive. Control of air-borne radioactive hazards is a constant problem.

Personnel Decontamination

Contamination of the hands is the most important personnel-contamination problem met in industrial practice. The first step is to wash the hands thoroughly using plenty of soap, scrubbing all parts of the hands, under the nails, and across the knuckles with a surgical scrub brush. If this is inadequate, one or another of the alternative procedures suggested in Table 18-9 may be tried. In general, the mildest treatment possible should be tried first and harsher treatments resorted to only when these fail. Occasionally localized areas elsewhere on the body become contaminated. These should be cleaned with a cotton swab using one of the formulations given in Table 18-9. Sequestering agents should not be used in most cases, as they tend to increase the rate of adsorption of the contaminant into the body from contaminated skin.

Clothing

On delivery to the laundry facility, all articles are first monitored and segregated into one of three categories: (1) no activity detectable; (2) low-level contamination for which appropriate standards are 2 mr/hr gamma, 9 mrad/hr beta, removable beta-gamma (wipe method) 4,000 disintegrations/min/12 in.², removable alpha 1 disintegration/min/12 in.², fixed alpha 50 disintegrations/min/in.²; and (3) high-level contamination, anything in excess of one of the low-level values.

Table 18-9. Formulas Used for Decontamination of Personnel

<i>Formula</i>	<i>Remarks</i>
1. Soap and shampoo	1. Preliminary wash. Scrub well
2. Abrasive soap (Lava)	2. Use with water and brush
3. 50% cornmeal plus 50% detergent (Tide) plus water to a paste	3. Scrub well with brush
4. Solid mixture of 30% detergent (Tide), 65% Calgon and 5% Carbose (carboxymethyl cellulose)	4. Prepare a 5% water solution of this mixture for spot cleaning with swabs or for use on the hands
5. 8% Carbose, 3% detergent, 1% Versene, 8% water. Homogenize into a cream	5. Use without additional water. Rub into skin 3 to 5 min and wash off with water. Use only in stubborn cases since Versene may increase the absorption of the radioelement
6. Mechanic's waterless hand-cleaning cream	6. Good for grease. Less irritating than scrubbing methods
7. Lanolin hand cream	7. Use between washes when skin becomes irritated from heavy scrubbing
8. Saturated solution of potassium permanganate followed by sodium bisulfite to remove the stain	8. Kills outer layers of skin. Should be done only under a doctor's supervision. Used only as a last-resort method

High-level articles are soaked in a mild decontamination solution (3 per cent citric acid or Versene), rinsed, and dried. If this does not reduce the level to that suitable for the laundry it is either retreated or discarded. **Uncontaminated articles** are laundered in an "uncontaminated" laundry. Articles in classes 2 and 3 are usually laundered in separate equipment.

Minor Incidents (Spills)

The following method is suggested for decontaminating nonporous surfaces affected by the spill of a radioactive solution. (Tompkins, P. C., O. M. Bizzell, and C. D. Watson, Practical Aspects of Surface Decontamination, *Nucleonics*, vol. 7, no. 2, p. 42, 1950.)

1. Tackle the job immediately. Promptness will usually be rewarded by an increase in the degree of decontamination achieved and a reduction in the effort required.

2. Determine the extent of the radiation hazard. The degree of protection required is dependent upon the amount and nature of the activity involved. Protective clothing including rubber gloves and coveralls should be worn for all decontamination work, and rubber overshoes should be used if the material has been

spilled on the floor. A respirator should be worn if atmospheric contamination is a possibility. The contaminated area should be monitored periodically during the cleaning process with a suitable survey instrument in order to determine the degree of decontamination being achieved and the radiation hazard remaining.

3. Confine the active solution to as small an area as possible. This may be accomplished with blotting paper or any other suitable material. This "first line of defense" should be included when assembling the equipment for any operation where a major spill is likely to occur.

4. Draw off or remove any remaining radioactive solution. This may be accomplished with additional blotting paper or with an adaptation of the transfer-pipette technique and will substantially reduce the radiation and retard further contamination of the surface.

5. Keep the contaminated area moist. If the affected area is permitted to become completely dry, the difficulty of decontamination may be increased by a factor of 10 or more. Also, a contaminated area that is permitted to evaporate to dryness becomes a more intense source of radiation.

6. Leach the contaminated area batchwise two or three times with a suitable decontamination reagent. The reagent should be left in contact with the contaminated area for about 3 min and then withdrawn in the same manner as the original spill. This operation, in addition to the rapid removal of most of the original radioactive solution, will usually eliminate more than 99 per cent of the contaminant.

7. Accomplish the remaining decontamination necessary with a scrub brush and a suitable reagent. Sufficient quantities of the reagent should be applied to the contaminated area to keep it moderately wet. Brushing may be accomplished with a scrub brush on a handle long enough to keep the operator beyond the range of permissible exposure. The affected area is scrubbed for a period of approximately 1 or 2 min, the reagent removed, the area rinsed, and the brush discarded. Efforts at decontamination should be continued stepwise with periodic monitoring until it is apparent that the particular reagent or procedure is contributing nothing further to removal of the contamination.

Major Incidents (Runaway Reactors)

Personal protection is the first requirement following a major incident. The area should be evacuated immediately. When all personnel are safe, survey teams completely equipped with high-range meters and full protective equipment can evaluate the resulting situation and mark the boundaries of the contaminated area, which should immediately be handled as a "high-radioactive" area. Control points should be established and subsequent entrance or exit from the area should be under close supervision.

Radiation protection through **dosage control** is the limiting factor until recovery has been completed. Proper operation requires the establishment of buffer zones where needed, a careful evaluation of time-motion-radiation field parameters, and full dosimetry for all personnel involved. The general steps taken in sequence are:

1. Evacuate.
2. Establish limits of exclusion area and stay times.

18-22 SURFACE CONTAMINATION AND DECONTAMINATION

3. Establish work-scheduling program on which to base control points and monitoring requirements, including buffer zones and health-physics centers.

4. Remove larger objects which are highly radioactive to reduce general radiation-field level.

5. Establish recovery schedule based on evaluation of the work schedule and available personnel.

6. Modify reclamation methods to provide remote control, shielding, or gross recovery. Suitable methods and their probable efficiencies for exterior areas such as roadways may be found in *Nucleonics*, vol. 9, no. 5, pp. C-12-C-15, November, 1951; and Army Corps of Engineers and Bureau of Yards and Docks, Radiological Recovery of Fixed Military Installations, NavDocks TP-PL-13, August, 1953, also Air Force TO 00-110A-8, Oct. 30, 1954.

Special equipment which proves useful under these conditions is:

1. Rescue breathing apparatus instead of masks or respirators.

2. Beta-gamma discriminating meter to locate surface-deposited contamination concentrations in the presence of a high-gamma radiation field.

3. Gamma pinhole camera to locate concentrations contributing to the gamma field, but which may not be superficially deposited.

4. High-range (0 to 100 r/hr) survey meters to get better measurements of higher intensities.

5. 0 to 10 r self-reading dosimeters.

SPECIAL DEFINITIONS

Contamination: Radioactive contamination may be defined as the undesired presence of radioactive materials in amounts that may be considered harmful to the health and safety of personnel or the validity of experiments or products.

Contamination of a surface: The deposition and attachment of radioactive materials to surfaces.

Decontamination of a surface: The removal of radioactive materials from the surface.

Contaminability of a surface: The tendency of a surface to hold radioactive materials with which it has come in contact.

Decontaminability of a surface: A measure of the extent to which deposited radionuclides can be removed from a surface.

General References

Army Corps of Engineers and Bureau of Yards and Docks, "Radiological Recovery of Fixed Military Installations," NavDocks TP-PL-13, August, 1953; also Air Force TO 00-110A-8, Oct. 30, 1954.

Eisenbud, Merrill, H. Blatz, and E. V. Barry, How Important Is Surface Contamination? *Nucleonics*, vol. 12, no. 8, p. 12, 1954.

Feldman, M. J., Advantages of Ultrasonic Cleaning, *Nucleonics*, vol. 12, no. 11, p. 65, 1954.

Gevantman, L. H., B. Singer, T. H. Shirasawa, and H. K. Chan, "Liquid Hold-up as a Measure of Contaminability" (to be published).

Krakaner, S., H. Byfield, F. Legler, and G. Failla, Review of Characteristics of Health Monitoring Instruments, Calibration of Survey Instruments by Plane Thin Beta Sources and Natural Uranium, *AEC Rept.* NYO-5642, Apr. 1, 1955.

Krakaner, S., "Calibration of Survey Instruments SII-17C, SIC-17D, and CD-V-700 in Terms of Dose Rate at Contact Using Plane Thin Sources (P^{32} ; Sr^{90} - Y^{90})," Radiological Research Laboratory, Columbia University, 1955.

Nucleonics, vol. 9, no. 5, pp. C-12-C-15, November, 1951.

Parker, G. W., and G. M. Herbert, Decontamination Characteristics of Porcelain Enamel, *Nucleonics*, vol. 12, no. 11, p. 72, 1954.

Terrill, J. G., "Surfaces and Finishes for Radioactive Laboratories," Research Correlation Conference No. 3, National Academy of Sciences, Washington, D.C., November, 1951.

Tompkins, P. C., and O. M. Bizzell, Working Surfaces for Radiochemical Laboratories, *Ind. Eng. Chem.*, vol. 42, p. 1469, 1950; vol. 42, p. 1475, 1950.

Tompkins, P. C., O. M. Bizzell, and C. D. Watson, Practical Aspects of Surface Decontamination, *Nucleonics*, vol. 7, no. 2, p. 42, 1950.

Tompkins, P. C., and H. A. Levy, Impact of Radioactivity on Chemical Laboratory Techniques and Design, *Ind. Eng. Chem.*, vol. 41, p. 228, 1949.

Watson, C. D., T. H. Handley, and G. A. West, Decontamination and Corrosion Resistance Properties of Selected Laboratory Surfaces, *Oak Ridge Natl. Lab. Rept.* 732, August, 1950.

Section 19

PHYSIOLOGICAL EFFECTS OF RADIATION

By

JAMES J. NICKSON, M.D., *Chairman, Department of Radiation Therapy, Memorial Center for Cancer and Allied Diseases, New York, New York.*

HARRY N. BANE, B.S., *Memorial Center for Cancer and Allied Diseases, New York, New York.*

Total Body Irradiation of Mammals	19-2
Maximum Permissible Dose	19-7
Skin Reactions	19-8
Radiation and Abnormal Blood Counts	19-10
Effects on the Eyes	19-11
Effects on Other Internal Organs	19-12
Effect on Fertility	19-12
Effects of Radiation on the Embryo (Fetus)	19-13
Carcinogenesis and Longevity	19-14
Genetic Change	19-16

PHYSIOLOGICAL EFFECTS OF RADIATION

James J. Nickson and Harry N. Bane

See page 19-17 for General References.

Since the discovery of X-rays in 1895, it has been observed that ionizing radiations have an **effect on biological systems**. The degree and type of biological effect will depend on several factors. The volume and the region of the organism that is irradiated are of importance in determining the response. This is true of cells (Zirkle, R. E., and W. Bloom, *Science*, vol. 117, no. 3045, 1953) as well as of mammals. In general, the degree of biological response will increase with increasing dose. However, the time over which the dose is administered is important. If the total dose is delivered over a long period of time (**protraction**), or if it is given in separate doses over a period of time (**fractionation**) the biological effect will be less, and in some cases there may be qualitative differences as well.

The degree and kind of response may depend on the type and energy of the incident radiation. The different quantitative effect of two types of radiation on the same biological system is called the **relative biological effectiveness (RBE)** of the two radiations. The RBE is defined, in general, as the ratio of the doses of two radiations that will produce the same biological effect. Specifically, the comparison is usually made with X- or gamma radiation.

Some forms of radiation damage may be seen immediately, but most mammalian effects are not seen for some time following exposure. This time varies from minutes to many years, depending upon the organ being observed.

Units. In this section, the units rad and rem are used and the use generally is interchangeable. Where the rad is used, it generally infers that the effects observed were a result of X- or gamma irradiation and the conversion to rems to include other types of radiation is not justified on the basis of observations to date. When the unit r (roentgen) is used, it generally means X-ray or gamma-radiation dose in air measured without backscatter.

TOTAL BODY IRRADIATION OF MAMMALS

The **acute total body irradiation of mammals** at doses above a few hundred rads will have some lethality, and doses of a few thousand rads will kill all exposed animals. Animals which have received total body irradiation show a characteristic median survival time which is a function of dose, and this is shown in Fig. 19-1 (Langham, W., *Radiation Research*, vol. 5, p. 404, 1956). Various types of death

are associated with different regions of the curve, and they are generally referred to as modes of death.

1. In general, total body irradiation doses above 10^4 rads results in death within hours. The same symptoms and mean survival time can be reproduced by irradiation of the head alone, and hence this mode has been called the **central-nervous-system death**.

2. Between 10^3 and about 10^4 rads, there is a dose invariant mode of death with a mean survival time of about 80 hr which occurs in all species and which can be duplicated by irradiation of the **gastrointestinal tract**. Damage to the lining of the intestinal tract is extensive and occurs rapidly. Death results from loss of water and mineral components of the body fluids from diarrhea. Replacement of water and minerals can permit survival of exposed animals from this mode of death, with subsequent death from modes which occur at doses less than 10^3 r.

3. There are probably several modes of acute death in the dose range of 300 to 1,000 r. The most studied of these is the death associated with the inhibition of the blood-forming organs, most particularly the bone marrow, and hence this mode of death is usually called the **bone-marrow death**.

The depression of the bone marrow results in a marked depression of white cells circulating in the blood (**leukopenia**). The lymphocytes drop to about one-half of the normal value within 1 to 3 days (**lymphopenia**), and the **granulocytes** decrease more slowly (**granulocytopenia**), reaching a minimum at about the fourteenth day following exposure. This drop in white cells reduces the organism's defenses against infections, while simultaneously bacteria are invading the blood from the intestine because of the radiation damage. The animal may die of infection (**septicemia**). The platelets, which contribute to the clotting mechanism of the blood, also decrease (**thrombocytopenia**), and death may result from hemorrhage. In this dose range some animals will survive without treatment.

A typical survival curve for Swiss-Webster mice in this dose range is shown in Fig. 19-2, in which per cent survival is plotted as a function of dose. The dose of radiation which will kill 50 per cent of the animals in 30 days is defined as the median lethal 30-day dose (LD-50/30).

Details of these modes of death are summarized in Table 19-1, which was first developed by H. Quastler. Each mode is characterized by its distinctive mean survival time t . The threshold dose that will produce death in the particular mode in 50 per cent of the animals is called the **median lethal dose LD-50** and is listed in column 3 of Table 19-1. D_{max} is the maximum dose that will produce a given mode

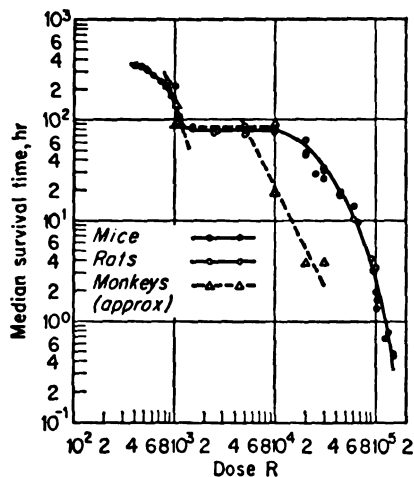


FIG. 19-1. Approximate median survival times of mice, rats, and monkeys after massive rapidly delivered doses of gamma radiation.

before another type of death with a shorter mean survival time occurs. Also tabulated are the species in which the various modes of death have been observed. This list is incomplete since careful investigation of causes of death at all doses has not been carried out with all species. In general, these various modes of death have been observed with all types of radiation. Certain of these types of death have been observed with all types of radiation. Some types of death may be duplicated by the irradiation of a single organ system, which is designated as a target region and listed in column 6. The site of the final phase of death may not be due directly to damage to the **target organ** but may result from secondary effects on the body as a whole from damage to the target organs. The site of final action is listed in column 7, and if the site is not known it is indicated by a question mark.

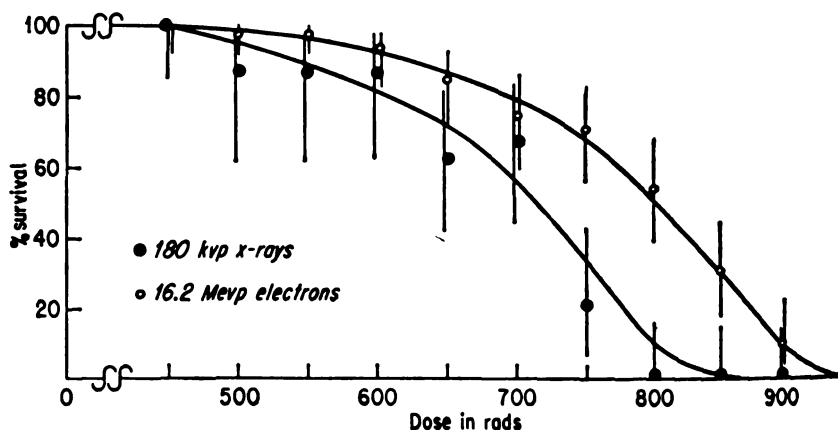


FIG. 19-2. Survival following total-body irradiation, X- and beta-radiation.

The characteristic symptomatology and physiology of all modes are not yet known completely, but some definite statements can be made and are listed in column 8. Column 9 shows the characteristic pathological findings. **Protective agents** exist which can increase the LD-50 or t if they are administered before irradiation. Reduction in the **oxygen tension** at the time of irradiation and the presence of **sulphydryl** containing amino acids offer some protection in all modes of death, but specific modifiers also exist which can increase survival in certain modes. **Bone-marrow injections** and **spleen transplants** after irradiation increase the LD-50 for the bone-marrow death.

In addition to these well-established modes of death, others have been reported but are not well documented as yet and have been omitted from Table 19-1.

Very little is known about the effects of **acute total-body irradiation in man**. Data from the exposed Japanese population of Hiroshima and Nagasaki are unreliable because of lack of accurate knowledge of dose and the superimposed effect of traumatic injuries and burns. Moreover, careful and detailed medical observations were difficult in the acute phase following exposure. The best data on normal individuals come from the accidental exposure of the Marshall Island population (Bond, V. P., E. P. Cronkhite, R. S. Farr, and H. H. Hechter, Hematological

Table 19-1. Modes of Death from Radiation

Name of mode (1)	t , days (2)	LD-50, krad (3)	$D_{\max}$ (4)	Species (5)	Target region (6)	Site of final phase (7)	Physiology and symptoms (8)	Characteristic pathology (9)	Modifier (protective)
Brain.....	0.1-10 ³ dose-dependent	1 and up	?	Most mammals (not in hamsters)	Brain	?	Convulsions, increased ventilation, ? respiratory alkalosis	Short t scattered dead cells, long t demyelination, acute massive necrosis ?	Anesthesia (early only)
Shock.....	0.1-1	0.1-1	?	Rabbit (not in mouse)	?	?	Low blood pressure	?	?
Intestine...	3½-4½	1	10 25	Most mammals, hamster, not always guinea pig (?), not in rabbit (?), not in salamander	Small bowel	Small bowel	Diarrhea, fluid and electrolyte loss, dehydration	Complete loss of intestinal epithelium	Shielding of part of intestine increased t by D_{50} only slightly
Oral.....	About 10	1-2	15	Mouse, dog (?), rat, hamster	Oral cavity, especially lower lip, esophagus (?)	Not mouth (?)	Some diarrhea, respiratory wheezing	?	?
Bone marrow	10-16 (depending on conditions)	About 0.5	1	Most mammals	Whole marrow	Blood stream	Bleeding (thrombocytopenia), diarrhea, low white blood count, bacterial invasion from intestine (septicemia)	Depletion of bone marrow, bacterial invasion of most organs	Antibiotics, shielding of part of bone marrow, spleen transplant, bone marrow infection

Table 19-2. Recommended Maximum Permissible Doses *

Source	r/day	r/week	rems/year
1902, Rollins	10		
1925, Mutscheller	0.2		
1925, Lewis	0.2		
1927, Advisory Board (Holland)	0.04		
1928, Barclay and Cox	0.17		
1931, Advisory Committee (U.S.)	0.2		
1936, Advisory Committee (U.S.)	0.1		
1950, ICRP		0.3	
1957, NCRP		...	5

* Taken from Stone, R. S. (*Radiology*, vol. 58, p. 639, 1952).

Changes in Human Beings Exposed to Fallout Radiations, *Radiation Research*, vol. 3, p. 215, 1955). The most pronounced changes in the formed elements of the blood are seen in Table 19-3. The changes can be seen at doses above 100 rads. The white-cell count falls, with the lymphocytes dropping by 50 per cent within

Table 19-3. Blood Changes

Dose/time	WBC	PMN	Lymphat.	RBC	Platelets
27 r/1 day	*	*	*	*	*
7 r/1 day $\times$ 3	*	*	*	*	*
60 r/1 day	Slight decrease	Slight decrease	Slight decrease	*	*
120 r/1 day	Moderate decrease	Slight decrease	Moderate decrease	*	*
150 r/1 day	Moderate decrease	Slight decrease	Sharp decrease	*	Sharp decrease
20 r/1 day $\times$ 15	Moderate decrease	Slight decrease	Moderate decrease	*	*

* No effects detected.

the first 3 days. Granulocytes and platelets begin to decrease about the nineteenth day following irradiation, and the maximum depression is reached at about the thirtieth day. At this time, there may be evidence of bleeding. There is a return to normal white counts by the sixtieth day. It should be noted that the time scale for blood changes in humans is different from that seen in lower animals where the decrease in the cellular elements of the blood starts promptly and is most marked at the fourteenth day and recovery is seen by the twenty-eighth day.

4. Animals surviving acute total-body irradiation and those receiving **sublethal doses** tend to die earlier than nonirradiated animals of the same strain. No characteristic pathology is associated with this phenomenon, and the animals seem to

die of the same diseases as nonirradiated ones. This has been termed the **acceleration of the aging processes, or life shortening**. Such animals do have a higher incidence of both benign and malignant tumors. Again, the types of tumors seen are those generally seen in a particular strain of animals, but the incidence is higher and they occur at an earlier age.

MAXIMUM PERMISSIBLE DOSE

As is evident from Table 19-2 the **maximum permissible dose** has been a constantly changing value. This has occurred in part because of changing concepts of the nature of radiation damage and in part because of increasing sensitivity of means of detecting radiation damage. Table 19-2, taken in part from R. S. Stone's article (*Radiology*, vol. 58, p. 639, 1952), shows the changing values until 1957.

Several types of injury may be produced by **occupational exposure**. These include:

1. Injury to skin
2. Injury to blood-forming tissue
3. Induction of malignant tumors
4. Other effects such as shortening of life span
5. Injury to genetic tissues

Of these, **skin damage** and **blood-forming organ damage** were described first. Within a few years after the discovery of ionizing radiation, skin injury was reported. In 1911, the first instances of leukemia in radiologists were reported. In 1925, Martland (*J. Am. Med. Assoc.*, vol. 85, p. 1769) described the first instances of **carcinogenic effects** of internally deposited isotopes (radium and mesothorium) in man.

As time went on, persons working for longer periods of time were observed. Initially skin cancers were common, especially of the hands. This fact led to increasing protection of the hands. Of the systemic effects **aplastic anemia** (failure of the bone marrow to make blood cells) was noted in the period 1910-1930. The occurrence of aplastic anemia in several persons after World War I focused attention on the need for lowering the occupational exposure. In 1927, the first official set limit was announced. By 1931, the U.S. Advisory Committee recommended 0.2 r/day as the **maximum permissible dose (MPD)**. Primarily because the use of harder (more penetrating) X-rays meant greater exposure to blood-forming tissues, in 1936 this value was halved to 0.1 rad/day. Experimental evidence developed during World War II focused attention on the histologic effects of amounts of radiation very little greater than 0.1 rad/day. This, in turn, led to the reduction of the MPD to 0.05 rad/day.

At the present time, the MPD level is set primarily by the possibility of **genetic effects**. As discussed in the section on genetics, there is much uncertainty about the amount of radiation which will produce detectable alteration in the spontaneous rates of mutation and the amount that we as a species can tolerate, but no uncertainty about the penalties of overoptimism about these values.

A new factor has been introduced into these considerations in the past few years. This relates to **nonoccupational exposure** from either industrial or military use of nuclear energy. This involves actually or potentially whole populations or sub-

stantial fractions thereof. Its possibility has brought to the fore the dangers of genetic damage. Genetically, 1 rad to the entire population is as harmful as 100 rads to 1 per cent of the population. This concern has resulted in the setting of MPD for populations as a whole, as well as reconsideration of the MPD for those occupationally exposed. The latter group, which even now is less than 1 per cent of the whole population, has increased by approximately a factor of 10 in the past decade and may be expected to increase further in the future.

Thus the MPD is not an absolute value, unchanged and unchanging. It is an educated guess at the amount of radiation that an individual or population may receive without detectable or, if detectable, significant harm. It has changed in the past as our ability to recognize damage has improved and as we became concerned about new types of damage.

Table of Occupational Exposure Levels

<i>Year</i>	<i>Rems/year</i>
1902	3,000
1931	60
1936	30
1950	15
1957	5

The population-exposure concept was first formally introduced in 1956-1957 with a recommended limit of a total, up to thirty years of age, or 14 rems, including natural irradiation (about 4 rems).

SKIN REACTIONS

Safe dose levels are not really known. It seems reasonably certain that, after 0.1 rad/day for 300 days/year, for 40 years, no changes would be detected.

Skin reactions are determined by injury to (1) blood vessels, (2) connective-tissue cells, and (3) epithelial cells. The interplay of injury to these three structures results in the clinical picture seen.

Exposure to high-intensity radiation of short duration (1 hr) in the range of 400 to 2,000 rads results in reddening of the skin (**erythema**) occurring initially within 2 weeks. In the lower dose range, no further reaction occurs. With higher doses, the skin may show dryness, loss of hair (**epidermitis**), or a moist reaction (**epithelitis**).

Effective means of treating existing damage have not yet been devised. This fact puts a great premium on avoiding damage.

Immediate radiation effects following exposure to high-intensity **short-duration** (1 sec to 1 hr) radiation (X-, beta, gamma):

400 to 1,000 rads—loss of hair at 2 weeks; returns, may be gray. Reddening (blood-vessel reaction), perhaps some browning (pigmentation) 1 to 3 weeks. No further reaction.

1,000 to 1,500 rads—reddening and browning 1 to 2 weeks, then dry scaling of superficial layers of skin (like peeling stages of sunburn). Onset of dry scaling about 3 weeks, lasting 1 to 2 weeks. Hair returns, usually gray.

1,500 rads +—reddening, dry scaling, then appearance of raw moist areas (as in severe sunburn) lasting 1 to 2 weeks with doses of 1,500 to 2,000 rads, longer with higher doses. With doses above 5,000 to 7,000 rads, raw areas may never heal. Hair does not return.

Late Effects—3 months or more:

400 to 1,000 rads—skin may return to normal appearance. May show slight tanning permanently. With higher doses, it may also show very small reddened areas because of dilated small blood vessels (**telangiectasia**).

1,000 to 1,500 rads—as above, except more severe. **Tanning** prominent. Skin thinned, dry. Easily injured by mechanical means. Heals slowly thereafter. Should be protected against excessive sunlight and against further irradiation.

Over 1,500 rads—protection against further radiation mandatory. Permanent skin damage at higher levels may be severe with repeated spontaneous breakdown. Surgical removal of the affected areas may be necessary. Danger of cancer greatest in this group.

Chronic Low-level Occupational Exposure

Statements in this section are only semiquantitative. Little or no data exist on amounts of radiation to skin received by persons who have subsequently shown damage. The reasons for this are many. The most important is the lack of general use of monitoring devices prior to 1940. Second is the difficulty in distinguishing rads from reps, especially with beta particles and soft X-rays. Known changes are:

1. Loss of detail **finger-ridge patterns**
2. Loss of hair (**epilation**)
3. Dryness of skin, nails
4. **Ridging nails**
5. Local overgrowth of small areas of skin (**keratosis**)
6. Delay in healing after mechanical or thermal injury—persistent ulcers
7. Development of skin cancer

These changes, once established, will last permanently. No further changes will occur except as below (**carcinogenesis**) or unless further radiation is received by that area of skin.

At 0.4 rad/day for 300 days a year, for 40 years, it is believed that certain changes might occur. These would consist of minimal changes, such as loss of detailed finger-ridge patterns and possibly loss of hair on fingers.

The amount of energy to produce acute effects varies with the energy of the radiation:

- 300 rads for P-32 beta rays
- 450 rads soft X-rays (100 kvp)
- 500 rads mid-range X-ray (250 kvp)
- 800 rads mid-range X-ray (500 kvp)
- 1,500 rads 2-Mev photons and Radium gamma

RADIATION AND ABNORMAL BLOOD COUNTS

In the earlier years of the use of ionizing radiations severe disturbances in **blood counts** as a result of occupational exposure were described. These involved diminished numbers of white-blood cells (**leukopenia**) and decreased numbers of red-blood cells (**anemia**). Several cases of severe and fatal cessation of formation of all the formed elements of the blood were reported (**aplastic anemia**). As a result several advisory bodies recommended that blood counts be done periodically. During World War II periodic blood counts were done on all workers of the Manhattan Project. Jacobson, responsible for the studies of the Metallurgical Laboratory of the University of Chicago, concludes: "the great variation in the steady state of the various hematological constituents of the peripheral blood from one individual to another, the inaccuracy of modern methods for measuring these constituents, and the individual physiological variation from hour to hour, day to day, and month to month, make it almost impossible to interpret minor fluctuations possibly due to acute or chronic radiation exposure."

The NCRP has issued the following statement on this subject (*Radiology*, vol. 63, no. 3, p. 428, September, 1954):

1. Provided that radiation monitoring of personnel, and where applicable of sites, is carried out by instruments (film, badges, pocket meters, etc.) in all circumstances involving potential exposure to penetrating ionizing radiations, **blood counts** should no longer be required as a method of monitoring.
2. Blood counts as a part of pre-employment, interval, and terminal examinations are good medical practice—to be done at the discretion of the medical officer in charge—but not as a part of a monitoring service.
3. Blood counts are a necessary part of the medical examination of anyone over-exposed to penetrating ionizing radiations.

It can be concluded that a routine blood count (red-blood count, hemoglobin, white-blood count, and differential) is of little value in the routine evaluation of occupational exposure at or under the MPD. Therefore, to advocate blood counts at intervals other than those associated with the routine annual physical examination now so common in American industry seems pointless. If, on the other hand, known over-exposure has occurred, the responsible physician will doubtless require frequent blood counts as a guide to management of the patients, as it is well known that depression of some or many of the formal elements of the circulating blood may occur under these conditions.

Patients have been studied by taking blood counts after total-body irradiation. Table 19-3 gives a summary of the findings. All histologically detectable effects, with the possible exception of slight permanent decrease in the platelet count, disappear after approximately 6 months.

The bilobed lymphocyte has been cited as a characteristic change in these cells in workers with cyclotrons. There seems to be some disagreement as to whether this is a common or reliable feature of lymphocytes in persons working with ionizing radiation, or in persons given total-body irradiation therapeutically.

EFFECTS ON THE EYES

The lens of the eye may be changed after exposure to ionizing radiations. This change consists of opacities in lens cells (cataracts). It starts equatorially in the posterior portions of the lens (Fig. 19-3). Cataracts from most other causes including spontaneous ones begin peripherally, e.g., near the edges of the longer axis of the lens (Fig. 19-4).

The changes may begin to appear as early as 6 months after exposure of the lens. There seems to be no upper time limit beyond which cataracts may not

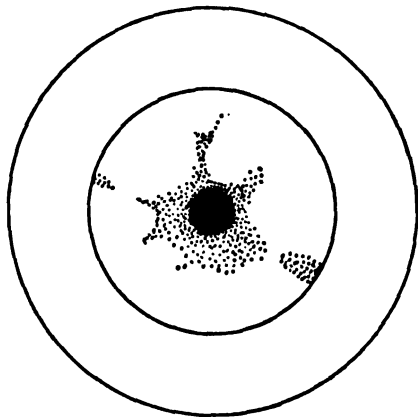


FIG. 19-3. Radiation cataract.

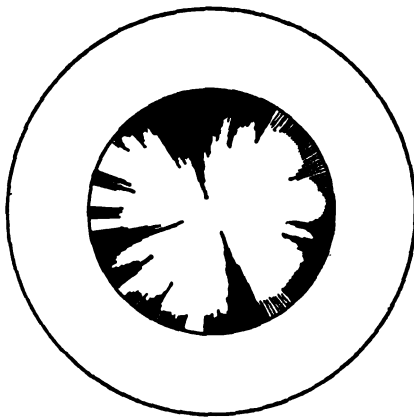


FIG. 19-4. Spontaneous cataract.

appear. The cells of the lens are injured more by the same amounts of energy left by neutrons than by X-rays. Fast neutrons from cyclotrons have accounted for the majority of the reported occupational cataracts.

Estimations of doses of fast neutrons to which the lenses of cyclotron workers were exposed varied from 8 to 270 reps. Estimation of threshold doses usually lies between 15 and 45 reps for men in early adult life.

Cataracts from occupational exposure to X-rays have not been recorded. It is not possible to give time-dose relationships for development of cataracts under these conditions. However, cataracts from X-rays under clinical conditions have been studied (Merriam and Focht, *Am. J. Roentgenol.*, vol. 67, p. 759, 1957). A single dose of 200 r (estimated) to a lens resulted in cataracts in two patients when they were first examined 19 and 22 years later. At the other extreme 1,100 to 1,200 r (estimated) was followed by a cataract 5.5 years later when the patient was first examined. Patients who received 175 r or less to the lens did not develop cataracts. With radiation treatment schedules over 3 months, the lowest dose associated with cataracts was 550 r. The first one was seen 3 years 8 months after treatment.

There seems to be an age dependency also. Children are more apt to develop cataract after a given dose-time exposure.

Other structures of the eye are comparatively resistant to late radiation changes. Acute changes occur only after high doses. These changes include inflammation of

the mucosa (conjunctiva) of the cornea and of the retina. These changes are seldom seen with doses of less than 500 r in a single exposure. Permanent damage, troublesome to the individual, would not be expected with doses of less than 1,000 r.

EFFECTS ON OTHER INTERNAL ORGANS

In general, as outlined above, the skin, in part because of its nearness to external sources of ionization, the gastrointestinal, and the blood-forming systems are the most sensitive to both the acute and the chronic effects of exposure to radiation. Doses which will produce either acute or late effects from total-body irradiation in other organ systems (respiratory, excretory, circulatory, nervous, or supportive tissues) are so high that death will have supervened from the effects on the blood-forming organs and on the digestive system.

However, it is possible that regions of the body may be irradiated to high doses (partial-body irradiation). Table 19-4 will serve as a guide to estimate approximately the dose levels that may be associated with the production of symptoms.

Table 19-4. Effects of Radiation on Various Critical Organs

Organ	Dose, rads	Effects	Symptoms of effects	Time of appearance after irradiation	Duration
Lungs	Approx 1,000 r	Swelling (?), ulceration	Cough, sob, dyspnea	4 weeks	2-4 weeks
U.R.T.	1,000 r	Ulceration	Soreness	3 weeks	2 weeks
Mouth	800-1,000 r	Ulceration	Soreness	2 weeks	2 weeks
Liver	500 r	?	Nausea, vomiting	24 hr	1-4 days
Pancreas . . .	2,000 r (est.)	?	?	?	?
Kidney	2,000 r	Vascular	Oliguria	4 weeks	Indefinite
Bladder	1,500 r	Swelling, ulceration	Frequency, urgency, pain	2 weeks	2 weeks
Muscle and heart	10,000 r (est.)	Swelling	?	1 day	Prolonged

EFFECT ON FERTILITY

In animals decrease in fertility has been demonstrated as an effect of exposure to ionizing radiation. Fertility may be measured by changes in litter size or in sperm count. Doses to produce changes in sperm counts in the dog are as little as 0.5 r if given daily for long periods of time. In mice, however, 1.1 r/day produces no detectable effect. Female mice of at least two strains (dba and C₃H) lived for five or more generations exposed continuously to 1.1 r/day without detectable effect on litter size. Females under the same conditions exposed to 4.4 r/day became sterile, however.

No data of this nature exist for man, since the size of families in our society is limited by many factors other than capacity.

For human beings, some data from women patients exist as to the amount of radiation to produce permanent cessation of ovarian function. The value is approximately 600 r if given in less than 1 week to women near the menopause. For women in the third decade of life, nearly 2,000 r in 1 week is necessary. No recorded instance of similar values for men exists. It is our belief that larger doses would be required to produce permanent sterility in the male.

Table of Effects on Fertility

Species	Dose	Effect
Mice, female....	1.1 r/day	None
Mice, female....	4.4 r/day	Sterility
Dog, male.....	0.1 r/day	Sperm-count reduction
Man (est.).....	1,000 r (single acute dose)	Sterility

EFFECTS OF RADIATION ON THE EMBRYO (FETUS)

Workers exposed to radiation often ask questions relating to the possible **effect of radiation on the fetus**. It will be evident from this discussion that the amount of radiation necessary to produce detectable effects upon the growing embryo is greater by an order of magnitude or so than those to which the worker is exposed. Nevertheless it seems worthwhile to give some background information to supply a basis for reply to questions about effects on the embryo.

The effects may be considered to be of two general kinds:

1. On developing cells of the *embryo*
2. On sex cells of the embryo—*genetic effects* (see Genetic Effects).

The effects discussed here will be those resulting directly from the first kind.

Effects on the fetus vary with dose and with the age of the fetus. The younger the fetus, the more sensitive it is, and the more damage results. L. B. Russell (*Radiology*, vol. 58, p. 369, 1952) showed that even 25 r produces detectable damage to the embryo of the mouse if given during the first half of fetal life. In man 1,200 r in air (approximately 400 r to fetus) will kill the fetus if given before the fourth month of gestation (Mayer, *Am. J. Obstet. Gynecol.*, vol. 32, p. 945, 1936). If radiation is given during the second half of the gestation period fetal death is less likely to occur. Instead, birth of a living but damaged stillborn infant may result. The most common human defects compatible with life have been microcephalic idiots (pin-heads). Anencephaly also has been common. Most of these are stillborn.

The Russells, in a series of papers, have shown clearly the association of damage with the system most actively growing at the time of irradiation. It is known that all systems—blood-forming, gastrointestinal, nervous, etc., do not grow uniformly throughout embryonic life. Some are most active at one stage of embryonic development, others at another. Dr. Russell's data show that the probability of abnormality produced by irradiation is greatest in the rapidly growing system. A

dose effect seems to be important in increasing the likelihood of producing damage, not in the kind of damage produced. If the dose is great enough, the fetus is killed, of course.

A major question is how little dose can produce an effect. No one knows for human embryos. L. B. Russell states that 25 to 50 r has produced detectable damage in mouse embryos. Correlation of mouse data to man is always a difficult question. It is the opinion of the authors that there is good reason to accept the data from lower mammals and assume that it applies to man, both as to the manner in which injury is produced and as to the dose necessary to produce it, at least as to the order of magnitude of the dose.

Since radiation can damage the developing fetus, and since the amount to produce a detectable increase is not known and defects similar to those produced by radiation can also be produced by other causes, in the opinion of the writers the safest occupational course is to remove all pregnant women from work with radiation. This is a stringent recommendation. It is made not because of the biological hazard, but because of the legal hazard. Inasmuch as the probability of an abnormal child normally is about 4 in 1,000 live births and since the kind of these abnormalities cannot be distinguished from those injured by irradiation of the growing embryo, it seems wiser to avoid the problem in so far as possible by removing known pregnant women from exposure to ionizing radiation. When the issue is settled in the courts the problem can be reevaluated. Also, when more is known as to the amounts of radiation necessary to double the spontaneous rate of abnormality, for example, perhaps relaxation may be possible.

CARCINOGENESIS AND LONGEVITY

Ionizing radiation can induce cancerous change (carcinogenesis) in experimental animals and in man. The amount of radiation necessary to produce a detectable change in the normal rate of tumor incidence in animals varies with the kind of cancer being studied. Table 19-5 shows some of the results from the literature and the kind of mammal used.

Table 19-5. Effects of Small Daily Doses in the Induction of Malignant Tumors*

r/day	Effect	Animal	Observer
0.11	Ovarian tumors, lung tumors	Mice	Lorenz
0.11	Mammary sarcoma	Mice	Lorenz
1.1	Mammary carcinoma	Mice	Lorenz
4.4	Leukemia (lymphomatous tumors)	Mice	Lorenz
1.1	Increased numbers of lung tumors	Guinea pigs	Lorenz
Up to 8.8	No induced leukemia, no ovarian tumors	Guinea pigs	Lorenz
1.1	Cancer of uterus with metastases	Rabbits	Lorenz
Up to 8.8	No ovarian tumors, no induced leukemia	Rabbits	Lorenz
0.1-1.0	Leukemia and lymphoblastoma	Rats	Barnett

* Stone, R. S., Concept of a Maximum Permissible Exposure, *Radiology*, vol. 58, p. 639, 1952.

In man, cancers of the skin, blood-forming organs, connective tissues, bone, liver, mucous membranes of the accessory nasal sinuses, and lung have been described as a result of occupational exposure (see Table 19-6).

Table 19-6. Cancers in Man

Location	External source	Internal source	Probable dose
Skin.....	X	..	?
Leukemia.....	X	..	?
Bone.....	X	X	4,000 rads
Mucous membranes.....	..	X	?
Lung.....	..	X	?
Connective tissue.....	..	X	?

The question of dose is really a question of dose and time. Animal studies show clearly that the incidence of cancers is a function of both the dose and of the time over which the dose acts. Increase of either enhances the probability that a cancer will develop. There is also a **latent period** of induction. No matter how large the dose, an irreducible minimal time must elapse before a cancer will appear.

There is no reason to suppose that a dose-time relationship, as well as a latent period, do not operate in human radiation carcinogenesis as well. Little is known concretely about dose-time relationship in man. The latent period is probably not less than 3 years, and more likely is nearer 5 years.

Leukemia in occupationally exposed persons has been recorded 37 times up to 1948 (Hempelmann, *Fallout Committee Rept.* 989, GPO, 1957) and in persons exposed to nuclear warfare 93 times (*Med. Research Council Rept.* CMD-9780, p. 34, 1956). The majority of these persons have had **myeloid leukemia**. It is believed that relatively few persons have had lymphatic leukemia as a result of exposure to ionizing radiations.

Cancer of the skin was common in the early days of radiology. Many radiologists had multiple amputations for cancers of the hands. As the need for protection of the hands became more widely appreciated, the numbers of new cases decreased. Today few skin cancers are seen as a result of occupational exposure in persons under sixty-five.

Bone cancers (**osteogenic sarcomas**) were initially described in radium-dial painters. The mode of entry of the radium was through the mouth, as a common practice was to tip with the lips the brush loaded with radioactive material. About 20 cases of bone cancers have been recorded as a result of occupational exposure.

Lung cancer has been found in miners in the Jachymov mountains. Radon gas in high concentration has been found in some of these mines. It is now believed that the radon gas and its decay products were responsible for the reported high incidence of cancer of the lung in this group.

Life shortening of experimental **animals** after exposure to ionizing radiation has been frequently described (Mole, R. H., *Nature*, vol. 180, p. 456, 1954). No specific injury is thought to be responsible. There appears to be an acceleration of

the process of natural aging, with the result that the irradiated animals "wear out" physiologically earlier than the unexposed controls.

Evidence of life shortening in man for a similar phenomenon has been reported by Warren (*J. Am. Med. Assoc.*, vol. 162, p. 464, 1956). He concluded that radiologists die on an average of 5.2 years before other physicians. A central problem in any such study is identification of a suitable control group. The conclusion has been challenged on the basis that the control population, the rest of physicians, are slightly older than the radiologists studied and hence would have a greater average age of death.

Hardin Jones (*Kaiser Foundation Med. Bull.*, vol. 4, pp. 329-341, 1956) has examined the problem also and concluded that 1 r total-body irradiation shortens life by approximately 15 days. The data are suggestive but not conclusive. Failla and McClement have suggested (*The Shortening of Life by Chronic Whole Body Irradiation, Radiology*, vol. 78, p. 6, 1957) that 1 r shortens life by approximately 1 day. The problem evidently needs further study.

GENETIC CHANGE

Excellent brief discussions of the general **genetic effects of radiation** for the interested reader are contained in the following reports: *The Biological Effects of Atomic Radiation, Natl. Acad. Sci. Rept.*, pp. 2-33, 1956; *Hazards to Man of Nuclear and Allied Radiations, Med. Research Council Rept.*, pp. 24-42, 1956; and *Effect of Radiation on Human Heredity, World Health Organization Rept.*, 1957.

Much discussion is current regarding the tolerable amounts of radiation from the genetic standpoint. Genes are the physical carriers of heritable characteristics. They are on chromosomes located in the nuclei of all living cells. They change (mutate) at a natural rate, which is low but not established for many genes in man. The rates that have been established lie between 8 and 95 mutations/ 10^6 population per generation. Exposure of the gonads to ionizing radiation increases the rate of occurrence of **genetic mutations**. At this time, these changes are **cumulative** and **irreparable**, a unique (and unfortunate) biological situation. Mutations are almost universally harmful. It has been estimated that more than 98 per cent of mutations will diminish the ability of the individual bearing them to cope with his environment, if they do not kill him outright.

It is now felt by the responsible policy-making agencies (NAS, BMC, ICRP) that the human race could tolerate without significant harm a doubling of the natural rate of occurrence of mutations. The amount of radiation in man necessary to produce a doubling (the **doubling dose**) is not known. Experimentally, the range of doses to double the spontaneous-mutation rate lie between 9 and 390 rads. Most data resulted from experiments with plants and *Drosophila melanogaster* (fruit fly). Two values have been gained from studies of the mouse and are each about 50 rads. Estimates based on data from animals and on our very limited data on the natural mutation rate in man indicate that the dose of ionizing radiation sufficient to double the rate of spontaneous mutations for most genes in man lies somewhere between 15 and 150 rads, and probably lies between 30 and 80 rads. The estimated figure of 50 is often used.

Since concern here is with contamination of the genic control of the character-

istics of the species and thus, ultimately, with the capacity of the species to survive, a few words of comment seem in order. Much uncertainty today obtains relating to this subject. First, all evidence points to the absence of a repair mechanism for this class of radiation injury. For many reasons this is a statement of exclusion for low doses and low dose rates. The possibility therefore exists that healing may exist after exposure to very low doses and dose rates. Second, it is theoretically possible that principles and methods of repairing genic damage may in time be found. Research in nucleic acid metabolism and in nucleoprotein structure and metabolism seems today most relevant to this goal. Third, there is much uncertainty for man as to the spontaneous or natural rate of mutation, as to the amount of radiation in man necessary to double that rate, and as to the degree of change in the natural rate that is tolerable. With time and work these uncertainties will diminish. It may be expected that with new information the MPD will change for occupational exposures and for exposures of much or all of the population. The degree and direction of change will depend upon the numbers derived in the areas outlined in this paragraph.

General References

- Zirkle, R. E., and W. Bloom, *Science*, vol. 117, p. 3045, 1953.
Langham, W., *Radiation Research*, vol. 5, p. 404, 1956.
Bond, V. P., E. P. Cronkhite, R. S. Farr, and H. H. Hechter, Hematological Changes in Human Beings Exposed to Fallout Radiations, *Radiation Research*, vol. 3, p. 215, 1955.
Stone, R. S., *Radiology*, vol. 58, p. 639, 1952.
Martland, J. *Am. Med. Assoc.*, vol. 85, p. 1769.
National Committee on Radiation Protection, *Radiology*, vol. 63, no. 3, p. 428, September, 1954.
Merriam and Focht, *Am. J. Roentgenol.*, vol. 77, p. 759, 1957.
Russell, L. B., *Radiology*, vol. 58, p. 369, 1952.
Mayer, *Am. J. Obstet. Gynecol.*, vol. 32, p. 945, 1936.
Hempelmann, *Fallout Committee Rept.* 989, GPO, 1957.
Mole, R. H., *Nature*, vol. 180, p. 456, 1954.
Warren, J. *Am. Med. Assoc.*, vol. 162, p. 464, 1956.
Jones, Hardin, *Kaiser Foundation Med. Bull.*, vol. 4, pp. 329-341, 1956.
Failla and McClement, The Shortening of Life by Chronic Whole Body Irradiation, *Radiology*, vol. 78, p. 6, 1957.
The Biological Effects of Atomic Radiation, *Natl. Acad. Sci. Rept.*, pp. 2-33, 1956.
Hazards to Man of Nuclear and Allied Radiations, *Med. Research Council Rept.*, pp. 24-42, 1956.
Effect of Radiation on Human Heredity, *World Health Organization Rept.*, 1957.

Section 20

**SAMPLING EQUIPMENT (DUST, GASES,
AND LIQUIDS)**

By

HARRY F. SCHULTE, *University of California, Los Alamos Scientific Laboratory,
Los Alamos, New Mexico.*

Air-borne Contaminants.....	20-2
Nature of Air-borne Contaminants.....	20-2
Purpose of Air Sampling.....	20-3
Air-sampling Equipment.....	20-3
Determination of Radioactive Gases in Air.....	20-12
Particle Size of Aerosols.....	20-14
Accessory Equipment.....	20-21
Sampling Techniques.....	20-25
Quantitation of Samples.....	20-26
Calculations.....	20-27
Stack Sampling.....	20-27
Liquid Sampling.....	20-29

SAMPLING EQUIPMENT (DUST, GASES, AND LIQUIDS)

Harry F. Schulte

AIR-BORNE CONTAMINANTS

Nature of Air-borne Contaminants

Radioactive **air-borne contaminants** may occur in the air in various forms, depending on the method of generation and dispersion of the material. The physical form is often very important in evaluating the potential health hazard. Definitions of some of these forms follow:

Aerosol is the name given to any dispersion of solid or liquid particles of microscopic size in gaseous media.

Dust is a loose term applied to solid particles capable of being suspended in air or other gases. Processes like grinding, crushing, cutting, and drilling produce dust particles of sizes ranging from the submicroscopic to the visible.

Fume consists of solid particles generated by condensation from the gaseous state generally after volatilization from melted substances and often accompanied by a chemical reaction such as oxidation.

Smoke consists of small gas-borne particles usually resulting from incomplete combustion consisting primarily of carbon and other combustible material. The individual particles may be liquid or solid.

Mist is a loose term applied to dispersions of liquid particles in a gas. It is usually formed by condensation of droplets on suitable nuclei or by atomization of liquids.

Fog is a type of visible mist usually formed by the condensation of water.

Smog is a combination of smoke and fog.

Gas mixture is a state of dispersion whereby the dispersed material is in units of molecular size. If the contaminant is a gas in the atmosphere, it is much more difficult to separate it or isolate it than it is if it is a solid or liquid.

Since radioactive contamination may appear in the atmosphere in any of these forms, a variety of sampling techniques must be at hand. A filter paper which is effective against dust will break down in the presence of a mist. For this reason, it is necessary to give careful consideration to the method or process by which the air-borne contamination is created, since it is this which usually determines the form of the material in the air.

Purpose of Air Sampling

In any discussion of equipment for measuring the concentration of air-borne radioactive material, it is necessary to consider the primary purposes of such measurements. From the standpoint of radiation hygiene there are two major purposes: (1) to assess potential health hazards to those breathing the air, and (2) to evaluate the effectiveness of control measures such as ventilation and enclosure. The primary objective, however, may be summarized as the attempt to evaluate the total quantity of aerosol taken into the lung during the working period or exposure period. While this is a somewhat oversimplified statement, it serves as a useful criterion for measuring the effectiveness of sampling equipment. On this basis it can be seen that the most important item of information is the concentration of the radioactive aerosol in the air. Particle size, since it determines how much of the material actually reaches the lung, is also important. Other factors that should be known are duration of exposure, solubility of the aerosol in body fluids, and chemical composition of the aerosol.

AIR-SAMPLING EQUIPMENT

The first step in measuring the concentration of a contaminant in the atmosphere is usually to separate out the material from a measured volume of air in some convenient form for chemical analysis or radioactive assay. This operation is usually termed **air sampling**, and a variety of techniques are available for it. Such methods as filtration, precipitation, impingement, and impaction can be used to sample for dusts, mists, fumes, and smokes. For gases special techniques must be used.

Basic experimental and theoretical data on **air filtration** have been published by Rodebush, Langmuir, Johnstone, Lapple, Davies, Ramskill, Chen,* and others. The operation is not a simple one since it involves a variety of mechanisms, one or more of which may be operative under a given set of conditions. These include separation by inertial effects, diffusion, electrostatic effects, direct interception, and others. Which effect predominates during the passage of an aerosol through a filter medium depends on the filtering velocity; particle size, shape, and density; percentage of voids in the filter medium; humidity; filter-fiber diameter, and other factors. For most filters inertia and diffusion are the principal factors to be considered. From the standpoint of **inertia**, the effectiveness of a filter in removing an

* Rodebush, W. H., Filtration of Aerosols, in "Handbook of Aerosols," p. 117, U.S. Atomic Energy Commission, Washington, D.C., 1950.

Langmuir, I., *OSRD Rept.* 865, 1942.

Johnstone, H. F., and M. H. Roberts, *Ind. Eng. Chem.*, vol. 41, p. 2417, 1949.

Lapple, C. E., Separation of Dusts and Mists, in Perry, J. H. (ed.), "Chemical Engineers' Handbook," 3d ed., p. 1013, McGraw-Hill, New York, 1950.

Davies, C. N., Fibrous Filters for Dust and Smoke, *Proc. Intern. Congr. Ind. Med.*, 9th Congr., London, p. 14, 1948.

Ramskill, E. A., and W. L. Anderson, The Inertial Mechanism in the Mechanical Filtration of Aerosols, *J. Colloid Sci.*, vol. 6, pp. 416-428, 1951.

Chen, C. Y., Filtration of Aerosols by Fibrous Media, *Chem. Revs.*, vol. 55, pp. 595-623, 1955.

20-4 SAMPLING EQUIPMENT (DUST, GASES, AND LIQUIDS)

aerosol increases with increased particle size, particle density, and filtering velocity. From the standpoint of diffusion, the filter efficiency decreases with an increase in these factors. Hence there is reason to expect that the curves relating filter efficiency with particle size and those relating filter efficiency with filtering velocity would show either a maximum or a minimum value. Actual data have been obtained, indicating the existence of minimum values in these curves in the submicron particle ranges.* All these factors must be considered in evaluating reported data on filter efficiency. However, many experimental curves do not show these minima because the range of particle sizes and filter velocities covered is not broad enough.

Table 20-1. Effect of Flow Rate on Pressure Drop* and DOP Smoke Penetration† for Various Air-sampling Media ‡

Flow rate, lin ft/min	CWS-6	HV 70		Hurlbut glass paper	Whatman Chemical filter papers			Membrane filters		MSA type S
		9 mil	18 mil		No. 4	No. 41	No. 44	HA	AA	
5	0.015¶	5.0	0.47	0.001	84	89	6.5	0.002	0.002	45
	0.67 §	1.1	1.2	1.05	0.5	0.35	7.7	5.4	2.2	0.3
10	0.023	5.0	0.53	0.001	81	84	4.5	0.002	0.002	50
	1.45	2.2	2.45	2.2	0.95	0.68	15.3	10.9	4.3	0.6
20	0.04	3.5	0.65	0.003	77	77	1.4	0.1	0.002	52
	2.9	4.6	4.9	4.4	1.95	1.35	28.6	21.6	8.5	1.05
28	0.057	2.0	0.69	0.005	73	75	0.5	0.15	0.1	52
	4.05	6.3	6.9	6.1	2.8	2.0	40	31	11.8	1.6
50	0.045	1.7	0.45	0.005	62	67	0.2	0.15	0.015	51
	7.5	9.4	13	10.8	5.5	3.8	71	59.6	24.5	3.0
100	0.037	0.2	0.1	0.005	25	44			0.02	45
	17.0	21.8	27	19.8	11.5	8.1			39	6.5
150	0.018	0.1	0.025	0.003	12	29				34
	25.5	34.5	38.2	32.5	18.1	12.5				10.8
200	0.01					15				28
	35					17.8				16.3

* Pressure drop in inches of water.

† DOP smoke penetration in per cent (dioctyl phthalate particles 0.3 micron diameter, 50 µg/liter of air).

‡ Data by W. J. Smith and N. F. Surprenant, AECU-3119, Arthur D. Little, Inc., Cambridge 42, Mass.

¶ Per cent penetration.

§ Pressure drop.

Filtration may be done using various types of filter media including filter paper; various crystals; packed fibers of glass, minerals, or plastics; sand or other materials. However, for most air-sampling work filter papers are used. In recent years, extensive tests have been performed on numerous filter papers in an attempt to get

* Johnstone, H. F., and M. H. Roberts, *Ind. Eng. Chem.*, vol. 41, p. 2417, 1949.

Ramskill, E. A., and W. L. Anderson, The Inertial Mechanism in the Mechanical Filtration of Aerosols, *J. Colloid Sci.*, vol. 6, pp. 416-428, 1951.

Chen, C. Y., Filtration of Aerosols by Fibrous Media, *Chem. Revs.*, vol. 55, pp. 595-623, 1955.

Thomas, J. J., and R. E. Yoder, Aerosol Size for Maximum Penetration through Fiberglass and Sand Filters, *AMA Arch. Ind. Health*, vol. 13, pp. 545-549, June, 1956.

Thomas, J. J., and R. E. Yoder, Aerosol Penetration through a Lead Shot Column, *AMA Arch. Ind. Health*, vol. 13, pp. 550-555, June, 1956.

usable efficiency data. Typical data on filter-paper efficiencies are given in Table 20-1. It must be emphasized that these data represent the observations of one group of workers under the specific conditions given in the table. Since efficiency is a function of aerosol particle size, shape, and density and filtering velocity, it is impossible to quote simple figures for efficiency for a given paper. Also, techniques of measuring filter efficiency are very variable, and hence there is not good agreement among the various investigators. Table 20-2 merely lists references where data on

Table 20-2. References to Efficiency of Various Filter Papers

Designation	Manufacturer	Reference	Usage
HV-70	Hollingsworth and Voss	1, 3, 4, 5	General, cannot be dissolved
CWS-6	Chemical Corps	1, 2, 3, 4	Same
MSA type S	MSA	1, 2	Low resistance, must be ashed before counting
Glass paper	MSA 1106-B	3, 4, 5 }	High strength and chemical resistance
Glass paper	Hurlburt X935-B	1, 5 }	Can be leached but not dissolved
Whatman No. 1	Balston, Ltd.	1, 5	Cheap, available, can be dissolved
Whatman No. 4	Balston, Ltd.	1	Cheap, available, can be dissolved
Whatman No. 40	Balston, Ltd.	1, 3, 4	Cheap, available, can be dissolved
Whatman No. 41	Balston, Ltd.	1, 2, 3, 4, 5	Variable quality, can be dissolved, widely used
Whatman No. 41-H	Balston, Ltd.	1	Higher resistance, stronger
Whatman No. 42	Balston, Ltd.	1, 5	Cheap, available, can be dissolved
Whatman No. 44	Balston, Ltd.	1	High efficiency
Whatman No. 50	Balston, Ltd.	1	Cheap, available
Whatman No. 52	Balston, Ltd.	5	Cheap, available
Millipore HA	Lovell Chemical Co.	1, 6	Dissolves in solvents, high resistance, low strength
Millipore AA	Lovell Chemical Co.	1, 6	Lower resistance than HA, fragile

¹ Smith, W. J., and N. F. Surprenant, Properties of Various Filtering Media for Atmospheric Dust Sampling, presented at meeting ASTM, Philadelphia, Pa., July 1, 1953.

² Adley, F. E., R. H. Scott, and W. E. Gill, A Study of Efficiencies and Pressure Drop Characteristics of Air Filtering Media, Document HW-28065, Hanford, Aug. 10, 1953.

³ Fitzgerald, J. J., and C. G. Detwiler, Collection Efficiency of Air Cleaning and Air Sampling Filter Media, *Am. Ind. Hyg. Assoc. Quart.*, vol. 16, pp. 122-130, June, 1955.

⁴ Fitzgerald, J. J., and C. G. Detwiler, Collection Efficiency of Air Cleaning and Air Sampling Filter Media in the Particle Size Range of 0.005 to 0.1 Micron, *AEC Rept. TID-4500*, Dec. 9, 1955.

⁵ Silverman, L., and P. LaTurre, Collecting Efficiencies of Filter Papers for Sampling Lead Fume, *A.M.A. Arch. Ind. Health*, vol. 11, pp. 243-250, 1955.

⁶ First, M. W., and L. Silverman, Air Sampling with Membrane Filters, *A.M.A. Arch. Ind. Hyg. and Occup. Med.*, vol. 7, pp. 1-11, January, 1953.

filter efficiency for a given aerosol under a given set of conditions can be found. The table also serves to list the most common filter papers now in use for air sampling. In practice, the selection of a paper often depends on the resistance which can be allowed, temperature and humidity of the air to be filtered, and similar factors. In general, papers of high filter efficiency are likely to be those of high resistance also.

Attempts have been made to standardize on a few papers, but these have been unsuccessful so far. A considerable variety of filters are needed because of the varied sampling conditions encountered. The **all-cellulose filter papers** such as the analytical filter papers have a low resistance to heat and moisture. However, they are cheap, usually available, and can be counted directly with low loss. They can be dissolved in reagents for analysis or separation of components, if necessary.

Special papers, **cellulose and asbestos filter papers** (CWS-6, AEC-1, HV-70), are somewhat more expensive. They are more resistant to heat and moisture and can be counted directly with a slightly higher loss. They cannot be dissolved in common reagents, and hence analysis or separation is difficult.

The **glass papers** (1106-B and X935-B) are expensive and somewhat variable in quality. They have a high strength even when wet and are highly resistant to most chemicals and to heat. They can be counted directly with low loss. However, they cannot be dissolved in ordinary reagents.

The **millipore filters** are true sieves consisting of a plastic membrane perforated with tiny conical holes. They are very efficient, fragile, and expensive and require special samplers for use. They are easily soluble in many organic solvents and are rendered completely transparent when a drop of microscope immersion oil is placed on them. This makes them extremely valuable for sampling if it is desired to view the collected dust under the microscope. A technique has been developed for transferring the collected material directly to an electron microscope.*

Tubes and columns loaded with **sand, lead shot, glass fibers**, and similar materials have been used under special circumstances.† The collected material must then be leached out, plated, and counted. Such units have been used for measuring particle size and making studies on the mechanism of filtration.

Accessory equipment required for sampling by filtration is discussed elsewhere in this chapter.

Alternating- and direct-current **electrostatic precipitators** have found some use in work with radioactive aerosols. For a-c units power is furnished directly from a high-voltage transformer such as is used for neon lights. This precipitator has been described and evaluated by Drinker.‡ The d-c precipitator produces less of the troublesome ozone gas and operates at a slightly lower voltage. A commercial model based on the work of Barnes and Penney¶ is available and is widely used in industrial-hygiene work (Fig. 20-1).

The central electrode consists of an aluminum cylinder which tapers to a blunt point at the end. A stiff wire extends about 3 in. from this point. The outer, or collector, electrode consists of a cylindrical aluminum tube. The aerosol is precipitated on the inner wall of this collector electrode or on a liner of paper, cellophane, or metal inside the tube. The status of electrostatic precipitation for air-sampling work has been reviewed by Barnes.§

* Kalmus, Ernest, Preparation of Aerosols for Electron Microscopy, *J. Appl. Phys.*, vol. 25, pp. 87-89, 1954.

† Thomas, J. J., and R. E. Yoder, Aerosol Size for Maximum Penetration through Fiberglass and Sand Filters, *AMA Arch. Ind. Health*, vol. 13, pp. 545-549, June, 1956.

Thomas, J. J., and R. E. Yoder, Aerosol Penetration through a Lead Shot Column, *AMA Arch. Ind. Health*, vol. 13, pp. 550-555, June, 1956.

‡ Drinker, P., and T. Hatch, "Industrial Dust," 2d ed., McGraw-Hill, New York, 1954.

¶ Barnes, E. C., and G. W. Penney, An Electrostatic Dust Weight Sampler, *J. Ind. Hyg. Toxicol.*, vol. 20, p. 259, 1938.

§ Barnes, E. C., Atmospheric Sampling by Electrostatic Precipitation, in McCabe, L. C. (ed.), "Air Pollution," pp. 547-555, McGraw-Hill, New York, 1952.

For radioactive aerosols the electrostatic precipitator has few advantages and hence is not widely used in this field. The equipment presents many maintenance problems, particularly where lengthy samples must be taken. If the sample is to be subjected to chemical manipulation rather than counted directly, the precipitator is a convenient method of collection, since it can be removed easily from the metal surface with little foreign material added to it. Counters have been built which permit a direct count of the activity in the material deposited inside the tube. The liners can be removed and flattened out for direct counting. Since the material is

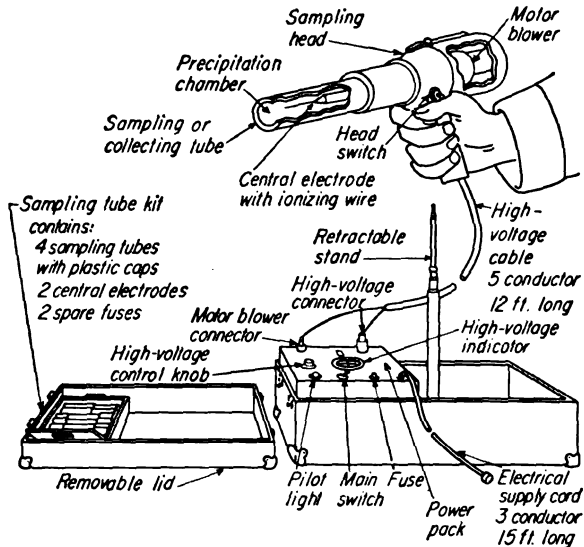


FIG. 20-1. Electrostatic precipitator. (From McCabe's "Air Pollution: Proceedings of the U.S. Technical Conference on Air Pollution," Fig. 8, p. 648, McGraw-Hill, 1962.)

deposited on the surface of the tube or liner, there is less loss on counting. Because of the hazards of high voltages on exposed parts, the electrostatic precipitator should not be allowed to operate for lengthy periods unattended. Electrostatic precipitators of either the a-c or d-c type are very efficient ($>98\%$), especially on fine dust. Hence they have found applications for fume sampling.

There is no real difference between the two terms **impingement** and **impaction**. However, the term **impinger** usually refers to a device in which an air stream passes through a jet and impinges on a plate or surface, both the jet and the plate being submerged in a liquid. It is essentially a device for collecting an aerosol in a liquid. **Impactors** are quite similar except that no liquid is used and the material is collected on the surface of the plate. The impingers are not very efficient for very fine dust and fume and hence are not widely used in radiation hygiene work. However, for sampling in hot wet atmospheres, impingers may be the only practical instrument available. Such conditions are encountered in sampling stack gases from incinerators, drying ovens, and similar operations. For higher efficiency the impinger may be followed by a glass-paper filter.

The material collected in the impinger is dissolved or suspended in the liquid medium and may be plated out for direct counting or microscopic examination. The latter is not possible, of course, if the aerosol is soluble in the medium. Two types of impingers are shown in Fig. 20-2.

Impaction devices of various types have been used for radioactive aerosols. Such devices are comparatively simple to build and operate and the collected material is usually in a convenient form for counting. The jet may be either circular or rectangular in cross section. The collecting plate is set at right angles to the center line of the jet, and the distance from jet to plate is the same order of magnitude as the dimensions of the jet.

Impactor collection efficiency increases with the particle size and with jet velocity. Sonic velocity is the limiting jet velocity, and at this velocity essentially all particles larger than 0.25 micron are impacted. However, at very high jet velocities, impacted material may be re-entrained and carried off the plate after being initially deposited. For this reason, the plate is usually coated with a sticky material to retain the particles. Glycerin, petroleum jelly, alkyd resins, silicones, and similar materials have been used for this purpose. The material must remain "tacky" under the conditions of use. The presence of this sticky material, if too thick, may interfere with direct microscopic examination of the collected material, with radioautography, and with direct counting of alpha-active materials.

In order to achieve the high jet velocities necessary for successful impaction small jet diameters and high-pressure drops are required. This limits the

amount of air which can be drawn through the device. As a result, impaction devices can be used only where air concentrations are relatively high or where long sampling periods are possible. An annular impactor developed by Tait* has been used for sampling for plutonium aerosols. The jet velocity is adjusted so that the plutonium particles are impacted and collected while the naturally radioactive material in the air is not collected. This is possible because the plutonium is chiefly in the form of large particles while the natural radioactive particles being

* Tait, G. W. C., Determining Concentration of Air-borne Plutonium Dust, *Nuclearonics*, vol. 14, no. 1, pp. 53-55, 1956.

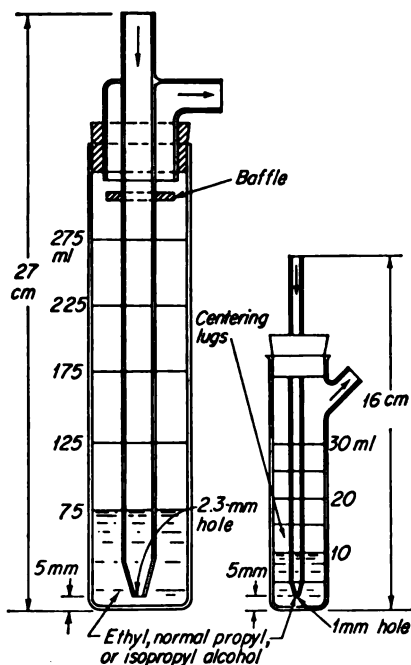


FIG. 20-2. Modified large impinger (left) and midjet impinger (right) for dust sampling. These instruments can also be used for gas, dust, and fume collection for certain contaminants. All-glass joints are also available. (From Silverman, "Industrial Air Sampling and Analysis," p. 48, Industrial Hygiene Foundation.)

formed from atoms of radon or thoron are very small. Experience with this device as reported by Hoy* has been very favorable.

Impaction theory has been discussed by May,† by Ranz and Wong,‡ and by Davies.§ They show that the efficiency of impaction is a function of the dimensionless parameter.

$$I = \frac{\rho V D^2}{\mu l}$$

where in May's terms I = dimensionless parameter

ρ = density of particles

V = jet velocity

D = particle diameter

μ = viscosity of air

l = diameter or width of jet

All geometrically similar systems having the same value of I will have the same impaction efficiency. The above authors give experimental data which partially confirm the theoretical treatment. For simple cases, it is possible to predict the efficiency of impaction for a given particle size from these data. An experimental curve for round and rectangular jets as given by Ranz and Wong is shown in Fig. 20-3.

Note that ψ is a slightly different form of I

$$\psi = \frac{C \rho_P V_0 D_P^2}{18 \mu D_c}$$

where C = correction factor for the resistances fluids oppose to the movement of small particles, dimensionless (for air at normal room temperatures and pressures $C = 1.00 + 0.16 \times 10^{-4}/D_P^2$)

ρ_P = density of aerosol particles, g/cu cm

V_0 = velocity of aerosol jet, cm/sec

D_P = effective diameter of aerosol particle, cm

μ = viscosity of air, poises

D_c = diameter of round jet or width of rectangular jet, cm

Because of limitations on sample size and difficulties of quantitation of impactor samples, these instruments are not usually used for routine air analysis. Impaction on a continuously moving plate or strip has been used in an attempt to obtain continuous records of air concentration. Most of these have failed because the sampling

* Hoy, J. E., and J. J. Croley, Annular Kinetic Impactor, Paper presented at American Industrial Hygiene Association meeting in St. Louis, Apr. 25, 1957.

† May, K. R., The Cascade Impactor: An Instrument for Sampling Coarse Aerosols, *J. Sci. Instr.*, vol. 22, pp. 187-195, 1945.

‡ Ranz, W. E., and J. B. Wong, Impaction of Dust and Smoke Particles on Surface and Body Collectors, *Ind. Eng. Chem.*, vol. 44, pp. 1371-1381, 1952.

Ranz, W. E., and J. B. Wong, Jet Impactors for Determining the Particle Size Distribution of Aerosols, *Arch. Ind. Health*, vol. 5, pp. 464-477, 1952.

§ Davies, C. N., *AMA Arch. Ind. Health*, vol. 4, pp. 354-397, 1951.

Davies, C. N., and M. Aylward, The Trajectories of Heavy Solid Particles in a Two Dimensional Jet of Ideal Fluid Impinging Normally upon a Plate, *Proc. Phys. Soc. London*, vol. B64, pp. 889-911, 1951.

20-10 SAMPLING EQUIPMENT (DUST, GASES, AND LIQUIDS)

volume is insufficient for radioactive dusts having low permissible concentrations. One type of impactor, the **cascade impactor**, is used regularly for measuring particle size rather than air concentration. It is discussed in the section on particle-size measurement.

When an aerosol passes near a hot surface, the aerosol particles are repelled from the hot surface because of the higher rate of bombardment by air molecules on the

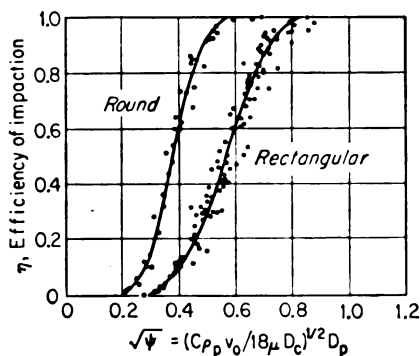


FIG. 20-3. Efficiency of impaction from round and rectangular jets. (From Ranz and Wong, *Ind. Eng. Chem.*, vol. 44, p. 1377, Fig. 8, June, 1952.)

hot side of the particles. Thus, in effect, there is a resultant diffusion force on the particle away from the hot surface. If the temperature gradient is made high enough this effect can be used to drive the particles to a cold collector. This is the principle of the **thermal precipitator**. While equations have been proposed for calculating the rate of diffusion, these have not been used in the design of existing instruments. These have been largely empirical in design.

Most thermal precipitators utilize an electrically heated wire as the hot surface. A thin glass plate in contact with a block of cold aluminum or brass serves as the cold collecting surface. Since a thermal gradient of about 4000°C/cm is

required for complete precipitation, the wire must be located very close to the cold plate. This then limits the sampling rate to a very low figure. The Casella instrument distributed in this country by Mines Safety Appliances Company samples at 4 ml/min. The original thermal precipitator of Green and Watson sampled at 7 ml/min.* Larger air-flow rates have been obtained by using a heated plate rather than a wire for the hot surface. Harrington† has described an instrument sampling at the rate of 1 liter/min. However, it is difficult to obtain even heating of the comparatively large surface area needed. Also, much higher temperatures are required, causing warping of the hot metal plate and consequent distortion of the air passage.

The thermal precipitator is an extremely efficient sampling device for very small particles. It is essentially 100 per cent efficient and has been used as a standard against which to measure other sampling instruments. It deposits the particles in a form which can readily be viewed in the microscope. However, there is a size separation, and hence it is necessary to measure particles in the entire deposit to obtain representative figures for particle size and concentration. Also, at high concentra-

* Green, H. L., and H. H. Watson, *Physical Methods for the Estimation of the Dust Hazard in Industry*, with Special Reference to the Occupation of Stonemason, Medical Council of the Privy Council, Special Reprint No. 199, H.M. Stationery Office, London, 1935.

† Harrington, E. R., and W. D. Crozier, *A Thermal Precipitator for Sparse Dispersions of Aerosols*, *New Mexico School of Mines, Rept. 2NR*, December, 1948, Project NR-092,013.

tions material is deposited in a thick layer directly under the wire. This interferes with microscopic examination and causes losses due to self-absorption if the deposit is counted for radioactivity. To avoid these conditions, oscillating thermal precipitators have been designed by Walton,* Laskin,† and others (Fig. 20-4). In these, the deposit is spread out into a ribbon with a uniform deposit over most of its width.

For radioactive aerosols, the thermal precipitator has largely served as a research instrument. It is particularly useful for collecting material to be viewed in the electron microscope.

One of the earliest devices for measuring dust concentration in air was the **settling chamber**, or **sedimentation chamber**. This consists merely of a boxlike device for trapping a fixed volume of air and allowing the particles in the box to settle to the bottom where they can be examined or counted in the microscope. Variations of this device have found use in radioactive-dust sampling. The simplest device is a simple **fall-out tray** laid in an area where a "dusty" condition prevails. There is a relation between fall-out rate and air concentration as follows:

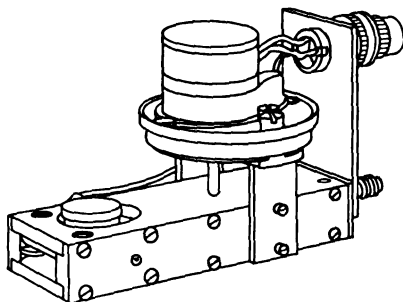


FIG. 20-4. Thermal precipitator.

$$C = \frac{R}{V}$$

where C = concentration, mg/cu cm

R = rate of fall-out, mg/sq cm/min

V = velocity of particle fall, cm/min

This assumes quiet air and constant replacement of the dust which settles out. In practice the fall-out tray is not usually used to measure air concentration but rather to measure the rate of accumulation of contamination on horizontal surfaces. With radioactive dusts this latter figure may be as important as air concentration since such surfaces become a source of external radiation.

To eliminate reentrainment of settled dust in cross drafts the tray may be filled with water or sand, or it may be replaced with sheets of gummed paper or simple metal sheets coated with a sticky material.

Useful data can be obtained by measuring the total activity collected on the tray. With a dry tray the collected material is brushed or washed off for counting. Water from the water-filled tray is evaporated before counting. The gummed paper is ashed. The sticky metal tray may be counted directly in a large counter, or the material may be washed off with a solvent and then ashed.

* Walton, W. H., R. C. Faust, and W. J. Harris, A Modified Thermal Precipitator for the Quantitative Sampling of Aerosols for Electron Microscopy, *Porton Tech. Paper* 1, ser. 83, March, 1947.

† Laskin, S., R. H. Wilson, and K. Lauterbach, An Oscillating Thermal Precipitator, "Encyclopedia of Instrumentation for Industrial Hygiene," University of Michigan, Ann Arbor, Mich., 1956.

20-12 SAMPLING EQUIPMENT (DUST, GASES, AND LIQUIDS)

The gummed paper or the sticky metal tray may be radioautographed with appropriate film. From the radioautograph, the activity of each particle may be estimated by measuring the spot diameter for beta and gamma emitters or counting tracks for alpha emitters. Skillern* has described a method for accurately sizing radioautograph spots using a high-contrast film to give sharp-edged spots.

Determination of Radioactive Gases in Air

The determination of gaseous radioactive contaminants presents many problems since it is generally not possible to designate a method which will work for all gases. In order to separate a contaminating gas from the air, it is necessary to take advantage of its specific chemical and physical properties, and hence the method will be different for each gas sampled. The one technique which is most generally applicable to gas is the introduction of a sample of the atmosphere to be tested into an ion chamber and the direct determination of the radioactive-gas concentra-

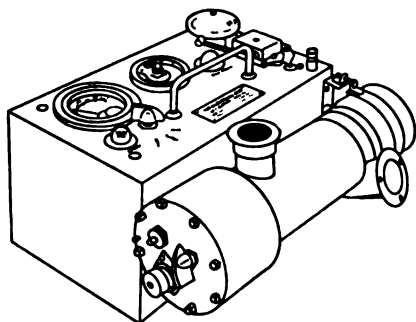


FIG. 20-5. Sniffer.

tion from the ion current produced. Tritium (H^3) can be determined in this manner by means of an instrument known as the "sniffer," which has been described by Eutsler.† The air to be sampled is continuously passed through an ion chamber of about 1 liter capacity. The resulting ion current is amplified and operates an indicator, recorder, or alarm. This instrument is only semi-quantitative since it measures the ions already present in the air which may be produced by radioactive sources, flames,

or other reactions. Even if all the ions are due to tritium, the number of ions present per curie of tritium depends on how long the tritium has been in contact with the air (Fig. 20-5).

These same defects apply to any direct-reading ion chamber. More quantitative results can be obtained by placing an ion trap ahead of the chamber to remove existing ions. This, of course, greatly reduces the sensitivity of the instrument for the gas being measured. The Kanne chamber is a large ion chamber of 16-liter capacity containing a deionizer to remove existing ions and particulate matter. It has been used extensively for the continuous measurement of radioactive gases in air, particularly tritium and the noble gases. A variation of these methods is to use an evacuated chamber which is opened in the atmosphere to be collected. The ion current is then measured by means of a vibrating-reed electrometer or similar device. In some cases, a Geiger tube rather than an ion chamber may be filled and the tube counted.

* Skillern, C. P., How to Measure Beta Activity of Fission Particles Using Film, *Nucleonics*, vol. 13, no. 12, pp. 54-56, December, 1955.

† Eutsler, B. C., G. L. Evans, R. D. Hiebert, R. N. Mitchell, M. C. Robbins, and R. J. Watts, Instruments for Monitoring Tritium in the Atmosphere, *Nucleonics*, vol. 14, no. 9, pp. 114-117, September, 1956.

Tritium may be determined by means of the "sniffer" or vibrating-reed electrometer and ion chamber. When it occurs in the form of water vapor as T_2O or HTO, these methods are also applicable. In addition, the tritium water may be concentrated by adsorption on silica gel or by bubbling the air through ordinary water. It can then be liberated from the silica gel by heat. Tritium in liquid water may be determined by liquid-scintillation counting.

The radioactive **noble gases**, i.e., neon, argon, krypton, and xenon, which are all beta and gamma emitters, may be determined in the Kanne chamber or other appropriate ion chambers. Since these usually occur as a complex mixture, the results will be obtained in terms of total microcuries per cubic centimeter of the mixture. As with any ion chamber, the instrument must be calibrated against the gases to be determined if possible. A reasonable calculation of the ion current for a Kanne chamber can be made from the relationship *

$$I = 9.57 \times 10^{-11} \frac{EC}{W}$$

where I = ion current, amp

E = average beta energy per disintegration, ev/disintegration

C = gas concentration, $\mu\text{c/cc}$

W = average energy required to form an ion pair, ev/ion pair

This formula holds for beta particles in the maximum energy range of 0.018 to 0.695 Mev.

Radon and thoron are alpha-emitting noble gases and are also usually determined by means of ion chambers counting the individual pulses. Radon may be determined in expired breath as well as in atmospheric air. In the former case, the purpose is to estimate the quantity of radium stored in a person's body. The method may be made somewhat more sensitive by adsorbing the radon from the breath by activated charcoal and releasing it into the ion chamber by heating the charcoal. This method has been described by Hursh.†

The determination of radon in air is now considered much less important from a health standpoint than it was a few years ago. The radiation dose to the respiratory tract in atmospheres containing radium decay products is now considered primarily due to the daughter products of radon. These, being solids, can be determined by filtration or other appropriate methods for particulate matter.

$S^{38}O_2$ and $C^{14}O_2$ are both beta-emitting gases and may be determined by ion-chamber methods after appropriate calibration. However, they are usually determined by chemical absorption, precipitation, and counting. $S^{38}O_2$ is absorbed in alkaline hydrogen peroxide using a frittered bubbler. It is then precipitated with barium chloride to yield $BaS^{38}O_4$. This latter can be filtered, plated out, and counted. An alternative method is to suspend the precipitated barium sulfate in a liquid-scintillation medium and count it in a liquid-scintillation counter. $C^{14}O_2$ is collected in alkali, precipitated as barium carbonate, and counted in the same manner as $BaS^{38}O_4$.

Iodine-131 frequently occurs as a beta-emitting vapor. It is usually absorbed in

* Fitzgerald, J. J., and B. W. Borelli, "Determination of Efficiency of Kanne Chamber for Detection of Radiogases," Document KAPL-1231, U.S. AEC.

† Hursh, J. B., The Measurement of Breath Radon by Charcoal Adsorption, *Univ. Rochester Atomic Energy Project Rept. UR-258*, 1953.

20-14 SAMPLING EQUIPMENT (DUST, GASES, AND LIQUIDS)

silver nitrate solution from which it precipitates as silver iodide. This, in turn, can be plated out on a planchet and counted or counted in a liquid-scintillation medium.

Frequent mention is made of the preliminary concentration of radioactive gases by adsorption on activated carbon, silica gel, or activated alumina or by condensation in a cold trap. Very few applications of such techniques to air-sampling methods have actually been used, however. Iodine-131 has been collected on charcoal and the gamma activity on the charcoal counted in a scintillation counter.

Particle Size of Aerosols

An important property of aerosols from the standpoint of hazard evaluation is their particle size. This factor determines the length of time an aerosol may be expected to remain suspended in the air as well as the depth of penetration into the lungs of a person breathing the air containing the aerosol. Particles larger than 10 microns in diameter, in general, settle rapidly and are caught in the upper respiratory tract rather than in the depths of the lungs. An equally important factor is the particle density, which should be considered along with the size. Since the factor of size is of most importance in connection with dusts and fumes, and since these are the more usual forms of atmospheric contamination, the following sections on particle size apply principally to dusts and fumes and to a lesser extent to mists.

The particle size of an aerosol is usually closely related to the method of **generating the aerosol**. Many air-borne radioactive-contamination problems arise as a result of **transporting powdered material** across an open room. The particle size which becomes air-borne is then essentially the same as that of the parent material. The highest dust concentrations are usually produced as a result of handling the finest powders. The powder being handled is not necessarily present in bulk form but may only be a film as an oxide coating on solid material.

Cutting of radioactive metal as with a circular saw, diamond saw, or abrasive wheel usually produces a fume of very fine oxide materials since most such materials at least partially burn during cutting. If the metal itself is not radioactive but merely contaminated with radioactive material, the particle size of the material released into the air will be characteristic of the nature of the contaminant.

Machining, especially if a coolant is used, usually does not produce significant amounts of air-borne material unless it is accompanied by burning of the material. In the latter case, the particle size is extremely small and characteristic of an oxide fume.

Grinding produces large or small particles, depending on whether it is coarse- or fine-grinding. Here, again, heating may cause formation of fine oxide fume.

Reference must be made to some of the texts cited * for a thorough treatment of

* Drinker, P., and T. Hatch, "Industrial Dust," 2d ed., McGraw-Hill, 1954.

Burnett, T. J., and T. Hatch, Estimating Air-borne Radioactive Particulate Hazards, *Am. Ind. Hyg. Quart.*, vol. 17, pp. 80-84, 1956.

Voegtlin, C., and H. C. Hodge (eds.), "Pharmacology and Toxicology of Uranium Compounds," National Nuclear Energy Series, div. VI, vol. 1, book 1, McGraw-Hill, 1949.

"Pathologic Effects of Atomic Radiation," National Academy of Sciences, National Research Council Publication 452, 1956.

this subject. However, there are some brief statements that should be kept in mind in evaluating the **biological effect of particle size**.

1. The smaller the particle the more deeply it penetrates into the depth of the lung. Larger particles are deposited in the upper respiratory tract and are removed from the body through various pulmonary actions to the throat where they are swallowed and pass through the gastrointestinal tract.

2. Extremely small particles tend to remain suspended in air even in the depths of the lungs and hence are exhaled with succeeding breaths. Thus, there is probably an optimum size for minimum retention in the alveoli of the lungs. There is considerable indirect experimental evidence to confirm this, but the difficulty of working with dusts of sizes smaller than 1 micron makes it hard to obtain direct evidence.

3. For a single particle of radioactive material, the amount of radioactivity of the particle is proportional to its mass or volume and hence to the cube of the particle diameter. Thus the radiation dose to the organ in which the particle lodges is much greater for a large than for a small particle.

4. For a given weight of material retained in the lung, the total surface area presented is much greater for small particles than for large, being inversely proportional to the square of the particle diameter. This large surface area usually means that small particles will dissolve more rapidly in the lung fluids and so pass into the blood stream. This is important in dealing with materials of low solubility such as uranium oxide.

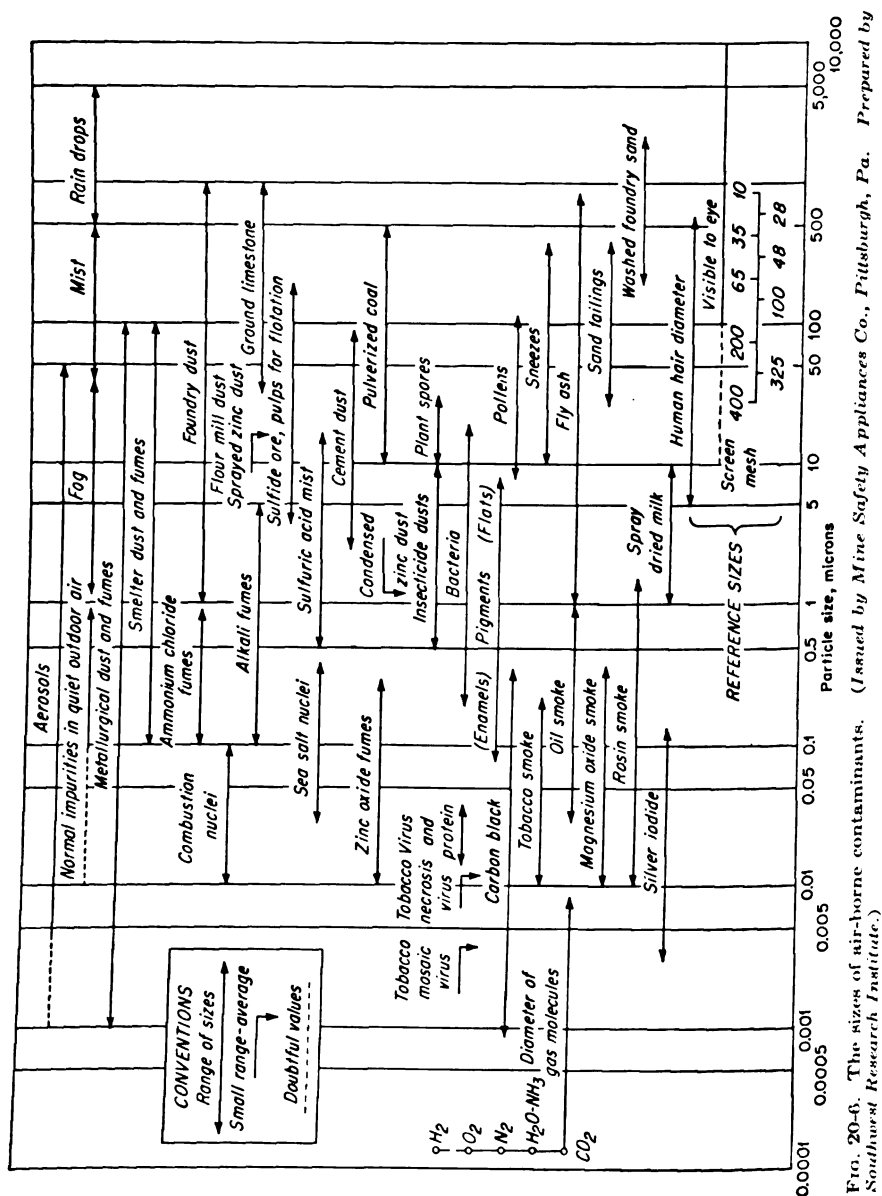
The chart shown in Fig. 20-6 shows the particle sizes of many representative dusts, fumes, and smokes. In order to utilize the information given by a particle-size analysis, it is necessary to have some method of **representing particle-size data** quantitatively. Practically no naturally occurring dusts or dusts produced by industrial operations occur in the form of particles of uniform size. Such dusts usually contain a fairly wide range of sizes and are said to be heterogeneous rather than homogeneous. Hence, in order to characterize the size of such dusts, it is necessary not only to designate the average size but also to include some measure of the size range or spread of sizes represented. Numerous studies of the size distribution of industrial dusts have led to a variety of means of representing these factors quantitatively.

Mathematically, there are a number of possible "average" sizes which may be used, depending on the purpose for which the average is intended. In practice the two most commonly used are the following:

The **number median diameter** is that size such that 50 per cent of the *particles* are smaller than it and 50 per cent are larger.

The **mass median diameter** is that size such that 50 per cent of the *weight* of the dust is of sizes smaller than the stated size and 50 per cent is larger.

A clear distinction must be made between these two types of averages whenever a size is specified, since the quantitative difference may be very great when the distribution contains a wide range of sizes. It would be difficult to designate the spread of the size distribution by a single term comparable with that used for the average size, if it were not for certain basic physical laws. It has been found experimentally that the sizes of most dusts are such that the *logarithm* of the diameter is distributed about the median according to the normal probability distribution. Since this



curve has only two parameters, one of which is the **geometric median size**, the other, the **geometric standard deviation**, or **sigma**, can be used as a measure of the size range.

Hence, after measuring the particle diameters of a large number (at least 200) of particles of a given dust, it is only necessary to plot the percentage of particles (or weight) less than a given size against the size to obtain a curve which characterizes the size properties of the given dusts. By using logarithmic-probability paper, this curve plots as a straight line and the two parameters can be obtained easily from this plot. This whole discussion is a much oversimplified version of what is a quite complex subject, and the interested reader should consult the excellent reference books on the subject.*

In all the preceding discussion, no attempt has been made to define the term **particle diameter**. Since most industrial dusts are not spherical in shape, the term "diameter" has a somewhat ambiguous meaning. It may be the longest particle dimension, the average of three dimensions, the distance between two extreme parts of a particle measured in a horizontal direction, the diameter of a sphere of equal volume, or some other statistical "diameter." So long as the same measurements are used consistently, the previous mathematical relationships will hold. Quantitatively, the differences between the various diameters are not great except when one dimension of the particle is very long or very short compared with the others, as in needle-shaped particles. Particle shape in itself may be an important property of the dust, especially in calculating specific surface areas or settling velocities (Fig. 20-7).

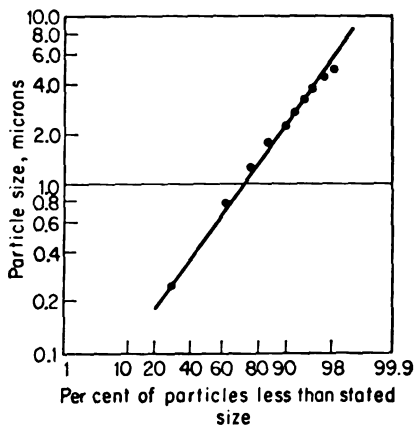


FIG. 20-7. Particle-size distribution of a typical uranium oxide dust.

The **settling velocities** of most dusts in still air are given by Stokes' law. For unusually large or small particles, settling velocities can be calculated using well-known corrections to Stokes' law. The usual form of **Stokes' law** for spherical particles is given by

$$V = \frac{d^2(\rho_1 - \rho_2)g}{18\mu}$$

where v = settling velocity, cm/sec

d = particle diameter, cm

ρ_1 = particle density, g/cm³

ρ_2 = air density

g = acceleration of gravity (980 cm/sec²)

μ = viscosity of air, poises

* Drinker, P., and T. Hatch, "Industrial Dust," 2d ed., McGraw-Hill, 1954.

Cadle, R. D., "Particle Size Determination," Interscience, 1956.

20-18 SAMPLING EQUIPMENT (DUST, GASES, AND LIQUIDS)

For sea-level air at 70°F this equation reduces to

$$v \text{ (cm/sec)} = 300,460 \rho d^2$$

or

$$v \text{ (ft/min)} = 592,000 \rho d^2$$

For nonspherical particles, which include most dust particles, an empirical shape factor must be introduced into the above equation. **Settling velocities for various particle sizes** are given in Table 20-3.

Table 20-3. Rates of Settling of Spherical Particles of Density 2.5

Size, microns	Settling velocity	
	ft/min	cm/sec
0.5	0.0037	0.0019
1.0	0.015	0.0075
5.0	0.37	0.19
10	1.48	0.75
50	37.0	19
100	148	75

Although it is often not appreciated, the **particle density** is a factor of considerable importance in characterizing a dust. The aerodynamic property of a particle which determines settling velocities, filter efficiencies, retention in the lung, and other factors is a function of ρd^2 where ρ is the particle density and d is the particle diameter.

It is important to remember that the density here is the density of the individual particle and not that of a loose mass of bulk dust. The latter contains a large volume of air and may be of much lower density than the individual particle.

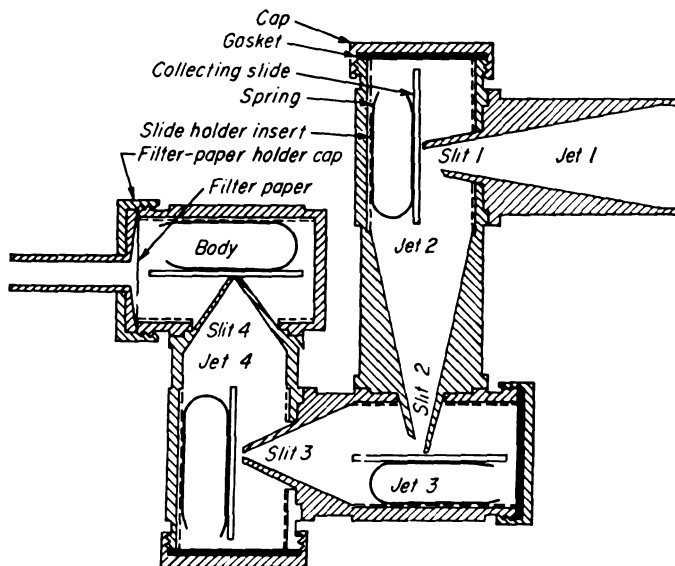
The obvious and most basic method of measuring the size of air-borne material is with the microscope. Techniques of doing this have been worked out and described in the literature.* However, these techniques are rarely applicable to the **measurement of particle sizes of radioactive aerosols** because the mass concentration of radioactive aerosol is usually extremely small compared with the concentration of nonradioactive dust which occurs in all air. There is no visual means of recognizing the radioactive particles. Under unusual conditions it is possible to single out the radioactive particles for visual examination by employing **particle autoradiography**, but the technique is so involved and time-consuming that it is seldom used. In this case, the aerosol is collected on a glass surface by some means such as impaction or settling, and then the surface is placed in contact with a nuclear emulsion or sensitive photographic film. After being held in contact with the film for a length of time, depending on the film sensitivity, the film is developed either while in contact with the aerosol or after removal from the aerosol-coated surface. After development, the film is again placed over the surface in exactly the original

* Drinker, P., and T. Hatch, "Industrial Dust," 2d ed., McGraw-Hill, 1954.

Cadle, R. D., "Particle Size Determination," Interscience, 1956.

position and the combination examined in the microscope with the emulsion side up. A radioactive particle is revealed by the presence of an exposure spot on the film. The particle causing the exposure should lie directly below the spot where it can be measured with the filar micrometer.*

Another method of measurement that is applicable to alpha-active dust is by **alpha-track counting** in which nuclear-track plates are placed in contact with a surface containing collected dust. After remaining in contact with the surface for a



Longitudinal section.

FIG. 20-8. The Cascade Impactor (Laskin). [From Voegtlin, Carl, and Harold C. Hodge (eds.), "Pharmacology and Toxicology of Uranium Compounds," vol. 1, p. 484, National Nuclear Energy Series, McGraw-Hill, 1949.]

known period of time (often 1 week) the plates are removed from the surface, developed, and examined in the microscope. The alpha tracks are visible, and star-shaped figures are seen where more than one track radiates from a single particle. By counting the number of tracks radiating from a given particle, it is possible to calculate the amount of radioactive matter in the particle. Then, assuming the particle contains only the radioactive material, and assuming a spherical shape to the particle, its diameter can be calculated.† A large number of particles must be measured because of the random nature of the disintegration process. This method, too, is very time-consuming and tedious.

* La Riviere, P. D., and S. K. Itchiki, Autoradiographic Method for Identifying Beta-active Particles in a Heterogeneous Mixture, *Nucleonics*, vol. 10, no. 9, p. 22, September, 1952.

† Leary, J. A., Particle Size Determination in Radioactive Aerosols by Radioautograph, *Los Alamos Sci. Lab. Rept. LAMS-905*, 1949.

The most common method of particle-size measurement for radioactive dust involves the use of the **cascade impactor** (Fig. 20-8). This instrument consists of four jets in series with jet sizes grading progressively smaller. After the fourth stage is a highly efficient filter. Air is drawn in through the largest jet where the larger particles are impacted and collected on a glass or metal plate. The air then passes through each of the jets in turn, depositing smaller and smaller particles, and finally through the filter where the rest of the particles are collected. The instrument is calibrated against an aerosol similar to that whose particle size is to be measured. For calibration a sample is collected, and, by optical means, the median size on each stage is measured. The instrument can then be used for measuring the particle size of a similar aerosol simply by collecting a sample and measuring the quantity of active material collected on each stage. Assuming that half the material on a

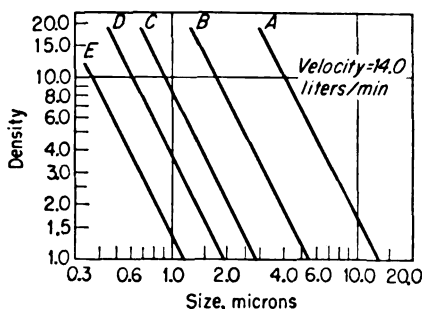


FIG. 20-9. Calibration curves. [Voegtlin, Carl, and Harold C. Hodge (eds.), "Pharmacology and Toxicology of Uranium Compounds," vol. 1, p. 500, National Nuclear Energy Series, McGraw-Hill, 1949.]

the stage median size, provided the instrument has been calibrated against another dust. All physical conditions of the impactor and sampling conditions being equal,

$$\rho_1 d_1^2 = \rho_2 d_2^2$$

where ρ_1 = density of dust particles for which impactor has been calibrated

d_1 = median size on a given stage for already calibrated dust

ρ_2 = density of dust particles for which calibration is desired

d_2 = median size on a given stage for dust for which calibration is desired

Laskin * has described the operation, construction, and calibration of the cascade impactor. Calibration curves for his impactor are shown in Fig. 20-9. Jet velocities for the impactors of May, Laskin, and Casella are shown in Table 20-4.

* Voegtlin, C., and H. C. Hodge (eds.), "Pharmacology and Toxicology of Uranium Compounds," National Nuclear Energy Series, div. VI, vol. 1, book 1, McGraw-Hill, 1949.

Table 20-4. Jet Velocity of Various Impactors*

Stage	May	Laskin	Casella
A	2.2	2.2	2.2
B	10.2	10.2	10.2
C	20.4	19.4	27.5
D	34.0	28.6	77.0

* May and Casella impactors operate at 17.5 liters/min; Laskin impactor operates at 14.0 liters/min.

Accessory Equipment

Air-sampling instruments are usually assemblages of sampling heads, tubing, pumps, and flowmeters. Only a few assembled units are available commercially. Mine Safety Appliances Company manufactures the electrostatic precipitator as a complete unit and also makes a high-volume air sampler. The Staplex Company makes a high-volume air sampler. However, for most air-sampling work it is necessary for the user to have some acquaintance with the individual units of equipment.

Pumps, blowers, and suction devices are used to draw air through every sampling device except the fall-out tray. The suction device may be simple **water displacement** where the sampling rate is quite low. This is a convenient suction source for the thermal precipitator, for example. Water is allowed to flow out of a filled reservoir and the air to replace the water is drawn in through the instrument.

For larger flow rates **electrically driven pumps** are usually used. Suction pumps available at scientific supply houses are convenient for this purpose. The Gast Manufacturing Company, Sutorbilt Company, Willson Products Company, and others have all built pumps which have been used as suction sources (see Figs. 20-14 and 20-15). These pumps are particularly useful when the pressure drop required is greater than 10 in. of water.

For still higher volumes household **tank-type vacuum cleaners** are much used. Care must be exercised in using these since many require a large flow of air to cool the motor and pump. If the sampling device restricts the flow of air excessively the motor may heat up and burn out. With a filter device, this heating condition may arise as a layer of dust accumulates on the filter and the flow is consequently reduced. Because of its high air-flow rate, the Electrolux cleaner has been extensively used for air-sampling work. The Filter Queen (Fig. 20-13), another tank-type cleaner, has been used frequently for sampling radioactive aerosols.

Where several samplers are to operate simultaneously from one suction source a large **commercial vacuum cleaner** may be used. For very large systems a unit similar to that on a **house vacuum-cleaning system** can be employed.

Where electricity is not available or where its use is obviated for any reason, **air ejectors** may be used. A suitable ejector for air sampling at 1 cu ft/min is made

20-22 SAMPLING EQUIPMENT (DUST, GASES, AND LIQUIDS)

by the Willson Products Company. An ejector may also be operated from a gas bottle containing carbon dioxide or freon gas. A **hand-operated pump** may be used at low flow rates where power is not available.

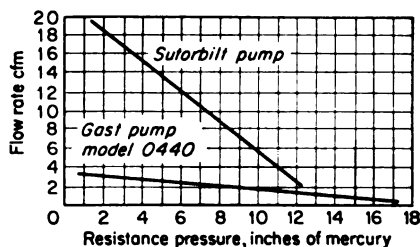


FIG. 20-10. Volume-pressure characteristics of two pumps.

to hold the filter paper in the air-sampling unit. These are as numerous as the various ingenious gadgeteers can devise. Such devices have been described by Silverman, Harris, Lauterbach, Burt, and others. One such device is illustrated in Fig. 20-11.

In principle these devices are simple, but there are several pitfalls that must be avoided by the designer. The contraction of flow on the downstream side must not be so abrupt that most of the air comes through the center of the filter. If a high-pressure drop is required across the filter, some form of "back-up" or supporting grid may be required to prevent tearing of the paper. The paper must seal at the edges so that all the air passes through the paper. The device should be opened and closed readily with a minimum of handling of the paper.

Flow-measuring equipment to measure the volume rate of air flow is a necessary part of the complete air sampler. The most commonly used device is the **rotameter**. These may be obtained in a very wide range of sizes. The other device commonly used is the **capillary flowmeter** which is a variation of the orifice meter and is illustrated in Fig. 20-12. A **sharp-edged orifice** may be used in place of the capillary in a similar system. The pressure drop across the capillary is related to the volume rate of flow and may be calibrated to read in terms of volume rate of flow. The capillaries may be made interchangeable so

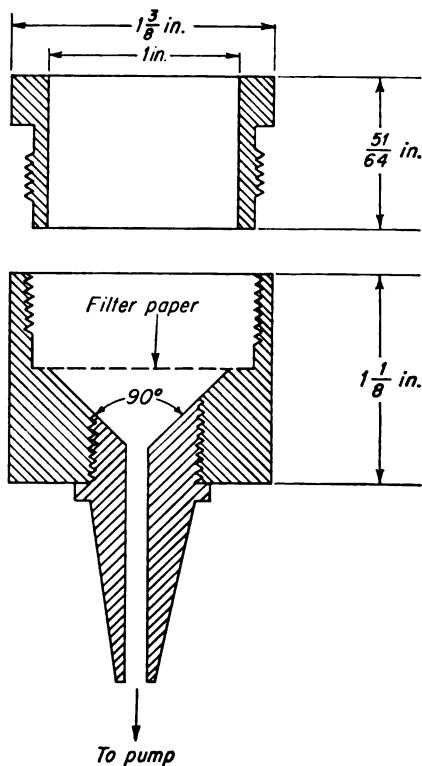


FIG. 20-11. Filter holder.

that a wide range of flow rates can be covered.

Both the rotameter and the capillary impose a resistance in the sampling head. This must be considered in the design of the instrument.

Any sampling instrument after being assembled must be calibrated. It may be calibrated against a wet test meter or a dry gas meter if these in turn have been calibrated against a standard. The best standard for calibration, however, is a large spirometer such as is used for calibrating household gas meters. All other flow-measuring devices are strongly affected by temperature and altitude changes. The wet and dry gas meters and the spirometer measure total volume of flow. The time interval must also be measured in using these devices for calibrating rotameters and capillary tubes which indicate rate of flow.

In the **assembled equipment**, which is the air sampler, ingenuity is again given free play. Most such equipment must be portable; hence, it is usually mounted on wheels or sliders. Three illustrations will show how the assembly may be made to serve the utility of the instrument, Figs. 20-13, 20-14, and 20-15.

Counting equipment is referred to here because it is an essential element in estimating air concentrations in the atmosphere. Details of counting are covered elsewhere in this volume. Counting equipment for air-analysis work differs from other standard counting equipment only in the design of the probe. This must be built to fit the filter paper, electrostatic precipitator tube, or impactor slide. The experienced designer of probes or chambers will have no difficulty meeting the requirements of the hygienist in this, but commercial equipment may not meet the requirements. In fact, the hygienist should consider chamber sizes of counting equipment before selecting the method of sampling or the filter-paper size to be used.

Both **scintillation counters** and **gas-flow proportional counters** have been used successfully for counting alpha emitters on filter paper. Beta emitters are also counted on similar equipment. Gamma emitters are usually counted with a

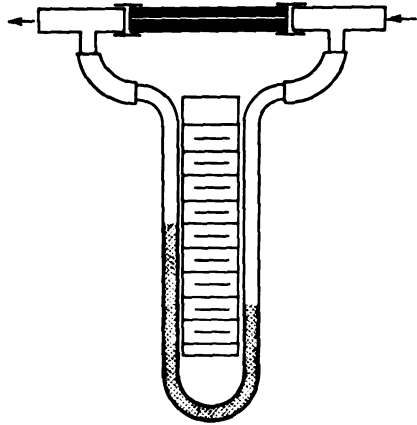


FIG. 20-12. Capillary flowmeter.

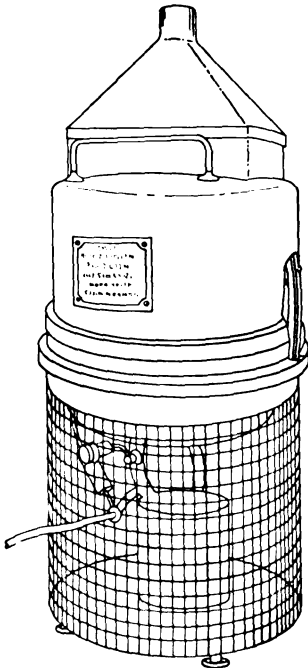


FIG. 20-13. Filter Queen air sampler.

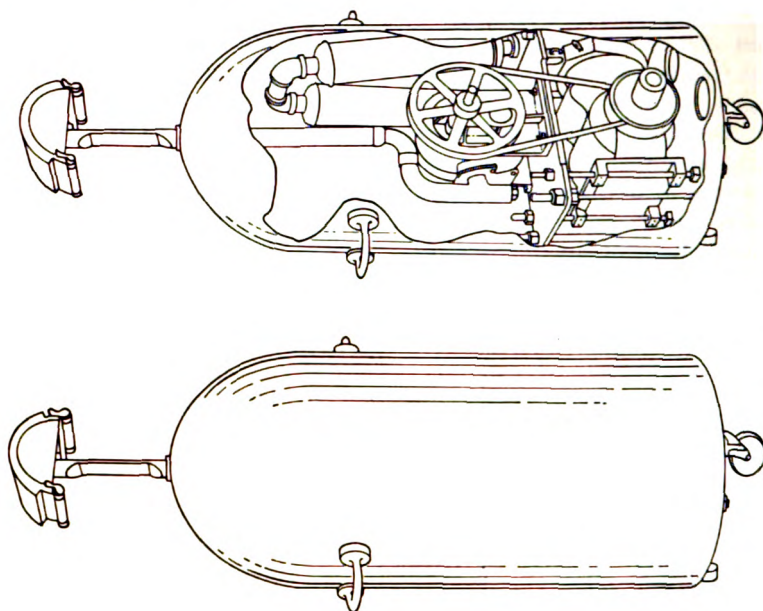


FIG. 20-15. Sutorbilt air sampler.

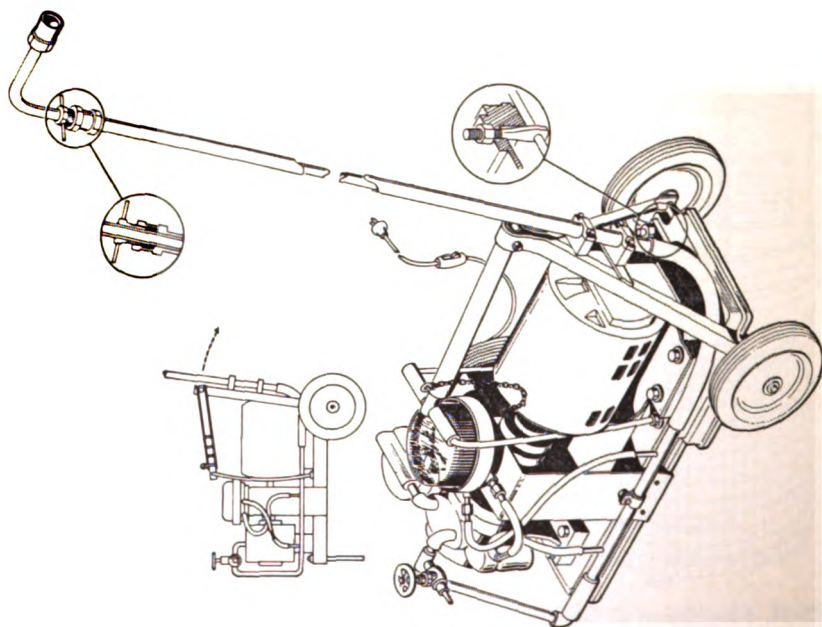


FIG. 20-14. Cast pump portable air sampler "Giraffe."

Geiger-tube instrument. Erratic results are sometimes obtained when counting high-level alpha- or beta-emitting samples on filter paper using a proportional counter with the filter paper inside the chamber. This is due to the fact that the paper, being an insulator, distorts the electric field inside the chamber, and this is particularly troublesome in regions of low humidity.

Combined sampler-counters are finding increasing use in this field. Because it is desirable to obtain immediate air-sampling results, if possible, many attempts have been made to combine the sampler with the counter and having the whole unit operate continuously. This may be done by using a moving strip of filter paper, a strip of metal foil for the electrostatic precipitator, or a rotating disk for an impaction device. The sample is then collected at one point and moves over under the counter at another point. The count rate is then applied to some type of recorder. To secure greater sensitivity, the filter-paper strip may move intermittently and carry the collected sample directly to the counter.

Such units are very appealing and useful, but all suffer from the handicap that the volume of air sampled is necessarily quite small if an immediate answer is to be obtained as desired. Where the permissible concentration is low, as in the case of plutonium, this is a very serious handicap. These units also do not discriminate between the short-lived natural activity in the air and that due to the contaminant. Some attempts have been made to utilize discriminator circuits to avoid this difficulty. This adds considerably to the electronic portion of the device, making it quite bulky. The large bulk of such sampler-counters in any case makes it difficult to use them for obtaining representative breathing-zone samples. One such unit is shown in Fig. 20-16.

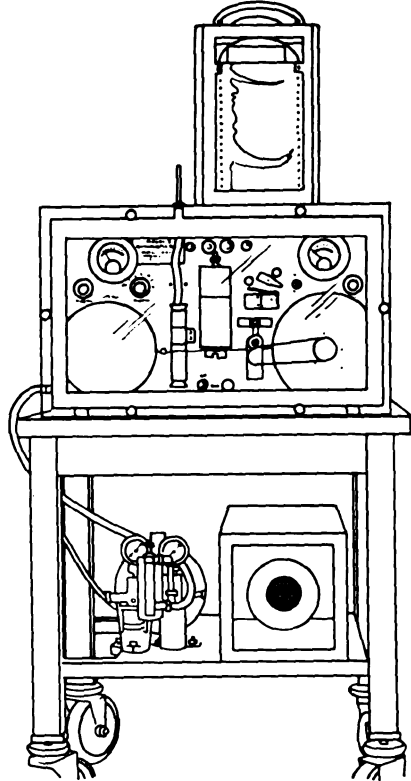


FIG. 20-16. Automatic sampler-counter.

SAMPLING TECHNIQUES

Two general types of samples may be distinguished: **breathing-zone** and **general-air samples**. Since the purpose of sampling is to determine the concentration of contaminant in the air which the worker breathes, a sample taken in the immediate vicinity of his nose is ideal for this purpose. It is practically impossible to take such samples during the entire working day and certainly not on repeated days.

20-26 SAMPLING EQUIPMENT (DUST, GASES, AND LIQUIDS)

Therefore, additional samples are usually taken by samplers located at various points in the workroom traversed by the worker. If the samplers are correctly placed and enough data are taken, and if a time schedule of the workers' activity in various locations is obtained, it is possible to assess the workers' average weighted exposure. Details of this have been discussed by Klevin and Harris.* This is a satisfactory technique to use when the operation being studied is repetitive at least daily. For nonrepetitive operations such as laboratory work, it is very difficult to obtain a measure of the workers' average exposure. Usually one or more samplers are located in the room where the operator works. The results on these samplers if correctly located are an "index" of the exposure. If very high concentrations prevail in one part of the room, the other samplers will register an increase in concentration. Sometimes it is possible to measure a true breathing-zone concentration for several days and relate this to concentrations measured on other fixed samplers in the room. This ratio then can be applied to subsequent results on the fixed samplers. In any case, the exact measurement of average exposure on laboratory workers leaves much to be desired. There is a real need for a small sampler which can be worn by the operator like a film badge.

Quantitation of Samples

After an air sample is collected and counted, corrections have to be made for **self-absorption** by the filter paper if this is the sampling method used. The dust particles penetrate a finite distance into the paper, and part of the activity is absorbed by the paper and not measured in the counter. This is particularly true if the substance is an alpha emitter. It is possible to obtain a factor for self-absorption by measuring the activity on the paper, then ashing it, plating out the remaining activity on a metal disk, and recounting it. Numerous measurements with Whatman No. 41 paper have indicated that only about 70 per cent of the alpha emitter in a paper is actually indicated by the counter. Similar data are needed on other papers but are not available. With millipore filters, electrostatic precipitator tubes, and cascade impactor slides, nearly 100 per cent is counted.

Another correction factor to be considered is that due to naturally occurring radioactive material in the atmosphere. This is formed from the disintegration of radon and thoron which diffuses from the earth's crust. The concentration in the air will vary greatly from place to place and from time to time. Hence, no fixed correction factor can be used. If no appreciable amount of thoron decay products are present, the collected sample can be allowed to decay for several hours and then counted. The **radon decay products** are short-lived and will decay out quickly. Whether this is possible should be determined experimentally. If substantial amounts of **thoron decay products** are present, the time for complete decay will be too long to be practical. In this case, it is best to count the sample after 4 hr and then recount it 24 hr later. The true count of long-lived alpha emitters then is given by

$$C = \frac{C_2 - C_1 e^{-\lambda \Delta t}}{1 - e^{-\lambda \Delta t}}$$

* Klevin, P. B., and W. B. Harris, Standard Procedures for Assessing Average Daily Air Contaminant Exposures, *AEC Rept.* NYO-4644.

where C_2 = count after 24 hr (t_2)

C_1 = count after 4 hr (t_1)

Δt = time $t_2 - t_1$

$$\lambda = \text{decay constant of thorium B} = \frac{0.693}{10.6} = 0.0653$$

If the measurements are made exactly at 4 hr and 24 hr then

$$\Delta t = 20 \quad \text{and} \quad e^{-\lambda \Delta t} = 0.271$$

Hence

$$C = \frac{C_2 - 0.271C_1}{0.729}$$

This, unfortunately, imposes a delay in obtaining concentration data. However, where a truly emergency situation exists the concentration is usually well above the natural background level and can be recognized.

Calculations

Concentrations of radioactive material in the air are usually expressed as microcuries per milliliter (cubic centimeter) or as disintegrations per minute per cubic meter, although other units may be used. Most counters have an **efficiency** or **geometry factor** which must be used in calculating the amount of activity on the paper. The essential calculation then is

$$\text{Conc.} = \frac{C}{EmR}$$

where Conc. = concentration in disintegrations/min/cu m

C = counts/min on paper

m = number of minutes of sampling

R = sampling rate, cu m/min

E = efficiency of counter

or

$$\text{Conc.} = \frac{C \times 10^{-12}}{2.22EmR}$$

where Conc. = concentration, $\mu\text{c}/\text{cu cm}$

All other symbols have the same meaning as before. This calculated concentration then must be corrected for self-absorption and natural activity, as previously discussed. Where desired, mass concentrations can be calculated from the above data. A table of the specific activities of the more common alpha emitters is attached for convenience in making calculations (Table 20-5).

Stack Sampling

In many cases it is desired to sample in stacks and ducts rather than in the open air. This imposes several complications not encountered in open-air sampling. Most air-sampling heads are too bulky to be inserted directly into the duct. Hence a tube of some sort should be used to carry the sample from the duct to the sampler. This should be as short and as straight as possible to prevent deposition of material in the tube. The end of the sampling tube should point upstream in the stack or

Table 20-5. Specific Activities of Various Alpha Emitters

Isotope	Half-life	Specific activity	
		Disintegrations/ min/ μ g	μ g/ μ c
Po ²¹⁰	138.3 d	1.0×10^{10}	2.22×10^{-4}
Ra ²²⁶	1,622 y	2.16×10^6	1.0
Th ²³⁰	8×10^4 y	4.31×10^4	51.5
Th ²³²	1.4×10^{10} y	0.245	9.1×10^6
U ²³⁴	2.5×10^5 y	13,700	165
U ²³⁵	7.1×10^8 y	3.82	5.8×10^5
U ²³⁸	4.5×10^9 y	0.742	3×10^6
Pu ²³⁹	2.4×10^4 y	1.38×10^5	16.1
Am ²⁴¹	500 y	6.6×10^6	0.33
Normal uranium		1.502	1.48×10^6
Normal thorium	Depends on metallurgical history of the metal		

duct so that it is not necessary to turn the particulate matter in the duct in order to draw it into the sampling tube.

In order to obtain a representative sample from a stack or duct, it is necessary

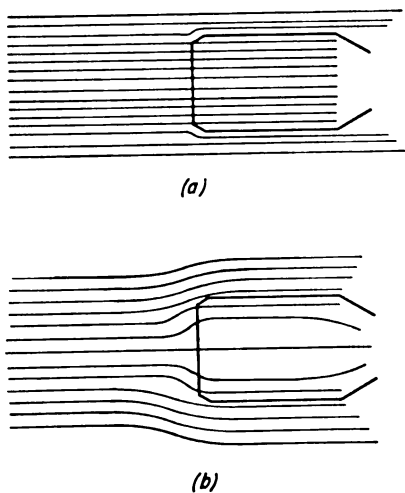


FIG. 20-17. Streamlines into sampling orifices. (a) Isokinetic sampling. (b) Sampling at half duct velocity. (Taken from photographs in article by May, J., *Sci. Instr.*, vol. 22, p. 187, October, 1945.)

that the sampling tube disturb the streamlines in the duct as little as possible. While it is obviously impossible to do this completely, a minimum of disturbance is created by **isokinetic sampling**; this is obtained when the sampling tube with a sharp edge faces upstream with suction applied at such a rate that the air velocities in the tube and the main air stream are equal. Figure 20-17 is taken from actual photographs by May * of streamlines made visible by smoke trails. Figure 20-17a shows streamlines into a sampling orifice during isokinetic sampling, while Fig. 20-17b shows similar streamlines when the sampling velocity is less than the duct velocity. Under conditions similar to Fig. 20-17b, or where the sampling velocity is greater than the duct velocity, an incorrect estimate of the concentration will be made by the sampler. Watson † has developed a theory of iso-

* May, K. R., The Cascade Impactor: An Instrument for Sampling Coarse Aerosols, *J. Sci. Instr.*, vol. 22, pp. 187-195, 1945.

† Watson, H. H., Errors Due to Anisokinetic Sampling of Aerosols, *Am. Ind. Hyg. Quart.*, vol. 15, pp. 21-25, 1954.

kinetic sampling from which Fig. 20-18 is derived. In this graph

- C = concentration measured
 C_0 = true concentration
 U_0 = stream velocity
 U = air velocity at entrance of sampler head

It can be seen that isokinetic sampling is important only for particle sizes larger than 5 microns. For large particles the error can get very large. If a sample is being collected for particle-size analysis, it is very important that the sampling be isokinetic, since otherwise the particle-size range collected will be different from that of the material in the duct (see Fig. 20-18).

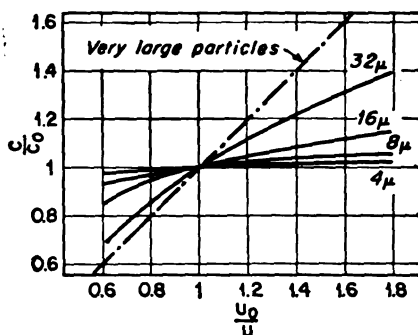


FIG. 20-18. Effect of nonisokinetic sampling on concentration measured as calculated by Watson. (Courtesy of the American Industrial Hygiene Association. From Magill, Holden, and Ackley, "Air Pollution Handbook," Fig. 10-2, p. 10-7, McGraw-Hill, 1958.)

LIQUID SAMPLING

Sampling of liquids for radioactive materials is a frequent requirement in many types of installations. Many of these are associated with the problem of waste disposal. The various waste liquids leaving the plant must be checked for radioactive content as they leave the area under control of the plant. In order to determine the source of such radioactive waste products, it may be necessary to sample in several liquid lines within the plant itself. The operation of a waste-treatment plant necessitates sampling of both influent and effluent streams as well as sampling at some intermediate points in the process.

The secondary loop of a reactor plant is regularly sampled for radioactive materials to detect leaks in the primary loop. Knowledge gained in this way permits correction of defects before they become serious. Public streams in the vicinity of an operating nuclear-process plant are usually checked for the presence of radioactive material as a precautionary measure. Such sampling should properly begin before the plant is built to provide a background against which later results can be measured. In the processing of radioactive liquids and solutions the problem of sampling is extremely important. It is quite possible to draw completely erroneous conclusions about a process if the samples drawn are not completely representative of the liquids being processed.

20-30 SAMPLING EQUIPMENT (DUST, GASES, AND LIQUIDS)

The general problems of liquid sampling differ considerably from those of air sampling.

1. The rate of mixing of liquids is much slower than that of gases. Hence, in many cases, it is desirable to sample immediately after a pump where mixing is very good. In other cases, it may be necessary to pass the material through a pump or mixer before sampling to ensure collection of a representative sample.

2. **Stratification** often occurs because of temperature differences. This is particularly the case in large tanks and ponds of still water. The composition of the liquid thus may vary vertically through the tank or pond.

3. Stratification may occur because of differences of specific gravity. Many of the liquids to be sampled are not true solutions but rather are suspensions. When the flow rate is low the nondissolved material tends to collect on the bottom. This is particularly serious where the contamination is due to some material such as plutonium which is readily adsorbed on finely divided solids.

4. The flow rates in streams, conduits, and pipes usually vary with time. This complicates the sampling problem since the concentration and flow rate are both varying with time.

The problem of **grab vs. composite samples** frequently arises in liquid sampling. The grab sample serves as a spot check and may be of very limited value since it represents a transient condition. On the other hand, it may be very useful in investigating an abnormal condition. A series of grab samples is very useful in tracing short-term variations in specific constituents. These are especially desirable when the plant operations involve batch dumping into waste streams.

Composite samples show average conditions and are very useful in computing quantities of radioactive materials discharged per day or per week. In many cases, the decision between grab and composite sampling must be made on a basis of the legal regulations governing rates of discharge into public streams and other bodies of water. If there is a maximum limit on the quantity which can be discharged in a short time interval then frequent grab samples must be taken. If the law regulates only the total quantity discharged per day then a composite sample can be used.

Sampling still water requires some method to ensure good mixing before collection of the sample. If the liquid is in a tank, it may be stirred or it may be circulated through a pump and back into the tank before sampling. If the liquid is in a pond, stirring or circulation is usually not possible. In this case a series of samples is usually taken at various depths and locations. These are then composited. The importance of such multiple sampling is emphasized by the phenomena of "turnover" of a pond. This occurs in cold weather or in early spring. If the water in the surface levels of the pond is cooled or warmed to 4°C, it becomes more dense than that below and the surface water sinks to the bottom while the water on the bottom comes to the top. Radioactive materials which may be in the sediment in the bottom layers of a pond are thus brought to the surface.

Sampling flowing streams or pipes presents many serious problems. The flow rate of the liquid in such streams usually varies with time. The samples may be collected at equal time intervals, with the sample volume being proportional to the total stream flow at the time of collection. Another method of sampling under varying flow conditions is to collect equal volumes of sample but to space the sampling period so that an equal volume of stream flow has occurred between

sampling periods. Either method can be accomplished by having a stream-flow indicator connected to the sampling device in such a way as to control the sample size or the sampling interval.

If a stream is to be sampled below the point of release of a contaminant, it is often possible to tell whether mixing is complete at the point of sampling by releasing a small stream of a dye solution and observing the mixing directly.

Liquid-sampling equipment is not standardized in any sense and is difficult to classify. **Manual sampling** is frequently done and may well be the most efficient when a study is being made over a limited time period. Manual sampling gives considerable flexibility to a program and provides a record of unusual occurrences which might affect the samples. For this work the sampling equipment involved is usually quite simple. A stainless-steel dipper or glass bottle and a stop watch are all that is required. With the aid of a stream-flow indicator, samples can be composited to take account of the varying stream-flow rate. Where stratification is suspected, it is desirable to collect samples throughout the full depth of the liquid. Various types of samplers have been devised for this purpose. They usually consist of a cylinder open at each end with provision for stoppering the lower end when the bottom of the stream or sewer is reached.

Mechanical sampling of liquids is used on permanent installations, and the wide variety of equipment used is a tribute to the ingenuity of engineers and technicians. One such sampler passes a container through the outlet of a V-notched weir at equal time intervals. The quantity of liquid collected during each passage is proportional to the stream flow through the weir outlet. An interval timer controls the frequency of sampling, and the individual samples are composited mechanically in a large bottle.

For sampling from a pipe which flows at a constant rate, solenoid valves which are controlled by an interval timer may allow the sample to flow by gravity into the compositing bottle.

Pumps may be used as sampling devices with a control mechanism connected to a stream-flow indicator so that the quantity pumped is always proportional to the total stream flow.

To achieve the kind of sampling desired, the variety of sampling devices used is almost unlimited. While a considerable number of liquid samplers are available commercially, many samplers are devised for the specific operation on which they are to be used. Sampling conditions are so varied that the best results are often obtained through an ingenious design using standard parts combined with parts made by the "gadgeteer."

Section 21

LIQUID AND SOLID WASTE DISPOSAL

By

ROLF ELIASSEN, Sc.D., *Professor of Sanitary Engineering, Massachusetts Institute of Technology, Cambridge, Massachusetts.*

ROBERT A. LAUDERDALE, Sc.D., *Associate Professor of Sanitary Engineering, University of Kentucky, Lexington, Kentucky.*

Philosophy of Waste Disposal.....	21-2
Site Selection.....	21-3
Liquid Wastes.....	21-8
Chemical Precipitation of Radioactive Wastes.....	21-13
Ion Exchange.....	21-20
Adsorption on Clay.....	21-23
Evaporation.....	21-27
Entrainment in Evaporators.....	21-32
Biological Processes.....	21-33
Disposal of Liquid Wastes.....	21-34
Discharge and Dilution.....	21-34
Storage of Radioactive Liquid Wastes.....	21-35
Ocean Disposal as Liquid or Solid.....	21-39
Disposal of Liquid Wastes to Pits and Wells.....	21-41
Fixation of High-level Wastes.....	21-46
Miscellaneous Methods of Disposal.....	21-49
Solid Wastes.....	21-50
Sources.....	21-50
Disposal.....	21-51
Incineration.....	21-54
Remelting of Contaminated Metals.....	21-58

LIQUID AND SOLID WASTE DISPOSAL

Rolf Eliassen and Robert A. Lauderdale

PHILOSOPHY OF WASTE DISPOSAL

See page 21-60 for General References.

The magnitude of the problem of radioactive-wastes disposal in this country—from a nuclear-power industry with potential capacities approaching 175 million kw by the year 1980—is such that the uncontrolled discharge of reactor fission-product wastes to the environment would be intolerable. By 1980, nuclear reactors in the United States will probably produce 1,500 lb of fission products per day. Solution of the challenging waste-disposal problems of this industry is an integral part of its development, particularly when one considers that the total activity of the wastes produced by 1980, even after 1 year of storage, will be on the order of 100 billion curies per year. This yearly production of fission products is roughly twice the quantity of radioactivity in all the earth's oceans in 1955.

Water is one of the nation's most valuable resources; water supplies relatively free from contamination are the life blood of municipalities and industries. Use of water by these will in turn lead to the production of waste products, many of which are carried by waste waters to streams and other public bodies of water. The doctrine of reasonable use of river water governs the acts of most regulatory agencies charged with the abatement of **stream pollution**. The user has a right to expect a reasonable quality of water from the stream at his plant and must in turn assure his downstream neighbor that he can secure water of reasonable quality from the same river. Treatment of liquid wastes by each group of users must be practiced so that the quality of water in the stream will not be unreasonably impaired through continued use and that the public health will not be in jeopardy.

The nuclear industry is bound by similar requirements and obligations. Basic policies have been formulated so that radioactive materials must either be entirely removed from nuclear-plant wastes or must be subjected to treatment or dilution so that the concentration of radioactivity will be reduced to a safe level before being released to the environment. Maximum permissible concentrations of radioisotopes in the environment have been reported elsewhere in this book. These must govern the waste-disposal policies of the nuclear-power industry, fuel-processing plants, research establishments, and all others discharging radioactive wastes.

In the past decade extremely dilute solutions of radioactive substances have been discharged to streams at such places as Hanford and Oak Ridge. Highly radioactive wastes from various plants of the Atomic Energy Commission have been

stored in tanks so that none of the "hot" material has been released to streams or the atmosphere. But if storage should be continued to be utilized, the accumulated solutions of wastes could amount to 200 million gal by 1980 and 2,400 million gal by the year 2000.

Better systems of disposal than storage are being developed through research. These include pit disposal in isolated areas, pumping through deep wells to salt domes, salt strata, or deep geologic troughs, or ocean disposal. Fixation of specific fission products on clays has also shown promise (Hatch, L. P., and W. H. Regan, Jr., *Processing of High-level Atomic Wastes with the View to Ultimate Disposal*, AEC Rept. TID-7517, pt. 1b, 1956). However, no system had reached the point of being an economic and engineering reality by 1959. All the disposal systems to be discussed later in this section will have to meet the requirements of the aforementioned policy of reasonable use of water resources.

The Atomic Energy Act of 1954 places more responsibility on industry to police its own waste discharges and maintain concentrations below tolerance levels. State and Federal health agencies are preparing personnel and legislation to enforce waste-disposal safeguards where radiation is involved. Radiation hygiene will become increasingly significant in health-department operations as legislation becomes enacted to define the basic philosophy of discharging only those wastes which will be innocuous to the public health. Sanitary-engineering divisions of state and Federal water-pollution control and health agencies will probably become the focal points of enforcement of this philosophy.

Safety precautions are of paramount importance in controlling the intentional or accidental release of radioactive wastes to the environment. Consideration must be given to the damaging effects of minute quantities of specific radioisotopes on living tissue. Thus, the approach to the disposal of these wastes must include control of the quantities released, knowledge of the behavior of individual radioisotopes from the point of release to the environment to the time of potential entry into a human body, where the physiological effects of these substances being discharged must also be known.

Eternal vigilance must be the watchword of the industry, because some of the radioisotopes most damaging to human tissue (Sr-90) have very long half-lives. This vigilance must include monitoring systems for routine discharges, as well as provisions to be discussed later for the retention of accidental discharges and prevention of their release to the environment in concentrations exceeding tolerance. For mixed radioisotopes such as mixed fission products the MPC value is only 10^{-7} $\mu\text{c/ml}$, or one-ten-billionth of a curie per liter. It does not take much imagination to realize the tremendous problem of safeguarding the environment to this low level when by 1980 the activity of fission products produced each year will approach 100 billion curies.

SITE SELECTION

As in the **selection of a site** for any industrial plant, the selection of a site for a nuclear reactor, a plant producing nuclear fuels from raw ores and reclaimed fissionable materials, plants for enriching fissionable material for use in nuclear fuels, chemical-processing plants for the separation of unspent fuel from fission

products, and research institutions will depend upon the availability of transportation, labor, housing, utilities, and other common factors. The site will also depend upon the nature of the wastes discharged, whether gaseous, liquid, or solid, and the character and concentration of the radioisotopes contained in these wastes.

The necessity of storing, treating, and disposing of wastes from normal daily operations is one of the most important considerations in selecting a site for a nuclear-energy plant. These factors will be discussed in detail, later in this section. The possibility of an extraordinary incident is another consideration which must be evaluated, with probability studies included on the frequency vs. degree of contamination basis for hazards running the scope from minor to major. Population densities in the immediate area of the proposed nuclear plant, as well as in the general region of the proposed site, even to a distance of 30 miles, must be included in the study of hazards of environmental significance (Gorman, A. E., Selection of Sites for Atomic Energy Plants, *J. San. Eng. Div., ASCE*, Paper 1168, February, 1957).

Water supplies must be studied in the light of the intended use of a plant. The volumes of water used for cooling purposes in operating nuclear reactors, nuclear-power plants, chemical-processing, and gaseous-diffusion plants are very large, running into millions of gallons daily. Flow rates for research reactors may be in the order of 10^3 to 10^4 gal/day. Water of good quality is required to keep induced radioactivity to a minimum.

No water supply is pure in the sense of being all water. The degree of impurity will depend upon the source of the water and its contamination by natural means, or by municipal and industrial pollution. Ground waters will usually contain more dissolved substances from the minerals with which they have been in contact, while

Table 21-1. Significant Water Constituents of Reactor Coolants from the Standpoint of Induced Radioactivity *

Constituent category	Energy category of neutrons			
	Thermal		Fast	
	Element or radical	Radioisotope produced	Element or radical	Radioisotope produced
Cations	Sodium	Na ²⁴		
	Arsenic	As ⁷⁶		
	Manganese	Mn ⁵⁶		
	Copper	Cu ⁶⁴		
Anions	Sulfate	S ³⁵ , P ³³	Sulfate	Si ³¹ , P ³²
	Chloride	Cl ³⁸	Chloride	S ³⁵ , P ³²
	Phosphate	P ³²	Phosphate	Si ³¹
	Chromate	Cr ⁵¹		
Dissolved gases	Argon	A ⁴¹		
	Chlorine	Cl ³⁶ , S ³⁵		

* Moeller, *AEC Rept. TID-7517*, 1956.

Table 21-2. Analyses of Some Larger Streams of the United States at High and Low Flows *

(Chemical analyses, in parts per million, except as indicated)

Date of collection	Mean discharge (cfs)	Silica (SiO ₂)	Iron (Fe)	Calcium (Ca)	Magnesium (Mg)	Sodium (Na)	Potassium (K)	Bicarbonate (HCO ₃)	Sulfate (SO ₄)	Chloride (Cl)	Fluoride (F)	Nitrate (NO ₃)	Dissolved solids (residue on evaporation at 180°C)	Hardness as CaCO ₃		Specific conductance (micro-mhos at 25°C)	pH	Color
														Calcium and magnesium	Non-carbonate			
Delaware River Near Narrowsburg, N.Y.																		
July 25-27, Aug. 7, 1949.....	333	12	0.02	17	5.0	5.1	3.3	40	29	6.5	0.0	6.2	97	63	30	157	7.1	10
Dec. 30-31, 1948.....	39,500	1.8	0.02	6.4	2.0	1.3		8	12	0.8	0.1	2.0	44	24	18	49.4	6.2	25
Savannah River at Augusta, Ga.																		
Aug. 15, 1950.....	3,490	14	0.09	3.0	1.2	4.6	2.0	2.2	2.0	3.1	0.2	0.4	40	12	0	43.4	6.7	3
Mar. 16, 1950.....	20,200	0.5	0.05	3.7	1.5	5.2	2.3	2.8	3.1	0.1	0.1	0.4	45	13	0	54.1	6.6	7
Ohio River at Ambridge, Pa.																		
Sept. 21-30, 1946.....	3,195	10	0.03	70	18	53	7.2	0	320	36	0.9	2.7	544	265	265	809	4.2	6
June 1-10, 1946.....	83,320	5.7	0.14	14	3.6	5.5	0.9	11	40	8.0	0.1	2.2	91	50	41	148	6.4	15
Kentucky River at Lock 8, Near Camp Nelson, Ky.																		
Aug. 17, 1949.....	665	11	0.05	22	6.0	9.8	37	9.0	0.2	1.5	0.4	2.2	59	80	31	209	7.7	7
Feb. 2, 1950.....	80,000	4.6	0.04	13	2.3	4.1	42	11	1.5	0.4	2.2	59	42	8	97.3	7.8	5	
Missouri River at Pierre, S. Dak.																		
Feb. 15, 1951.....	11,700	12	0.04	70	23	65	3.5	218	218	11	0.6	1.9	524	269	90	782	7.9	9
Apr. 8, 1951.....	106,000	13	0.04	46	13	36	3.7	174	91	4.0	0.3	4.8	302	169	26	455	7.8	9
Arkansas River at Ralston, Okla.																		
May 1-6, 1950.....	1,345			122	41	477	165	255	785			3.0	1,760	473	338	3,120		
July 18-20, 1950.....	55,200			37	6.4	31	108	18	55			2.3	254	119	30	400		
Bighorn River at Thermopolis, Wyo.																		
Apr. 13, 1950.....	830	22	0.04	84	23	114	212	330	21	0.4	0.4	4.2	706	304	130	1,040	7.8	
June 26, 1950.....	9,100	13	0.04	28	5.4	17	88	50	2.5	0.2	0.2	0.8	166	92	20	255	7.5	
Columbia River at Grand Coulee Dam, Wash.																		
Dec. 11-20, 1950.....	59,450	6.9	0.03	21	4.7	1.7	2.0	78	13		0.2	0.7	91	72	8	151	7.5	15
May 21-31, 1951.....	345,300	9.5	9.04	20	1.2	2.0	1.6	73	11	0.0	0.2	0.6	85	67	7	141	7.3	15
Colorado River at Lees Ferry, Ariz.																		
Sept. 11-18, 20, 1950.....	4,097	9.6	0.01	121	53	149	5.2	197	511	104	0.3	4.0	1,050	520	358	1,540	7.6	
June 11-20, 1950.....	53,220	11	0.04	44	13	23	3.4	140	83	14	0.3	1.1	262	164	49	430	7.5	
Sacramento River at Knights Landing, Calif.																		
Mar. 2-10, 1951.....	5,007	26	0.04	15	8.9	16	2.7	94	19	11	0.1	0.4	142	74	0	219	7.7	6
Apr. 21-30, 1951.....	19,810	24	0.08	14	6.4	8.8	3.6	77	12	6.2	0.3	0.6	103	61	0	157	7.6	10

* Love, S. K., and W. F. White, Analyses of Some Larger Streams of the United States at High and Low Flows, *Ind. Eng. Chem.*, vol. 48, p. 1248, 1956.

surface waters may contain a higher proportion of colloidal and dissolved solids. Water treatment may be necessary and may play a considerable part in plant economics. If these waters are to be used for reactor-cooling purposes, or in contact with neutron fluxes in the order of 10^{11} cm²/sec, induced radioactivity may result. Moeller (Moeller, D. W., *Reactor Cooling Water—Theoretical Evaluation of Induced Radioactivities*, AEC Rept. TID-7517, October, 1956) has listed the natural constituents of water shown in Table 21-1 as being important from the standpoint of fast- or thermal-neutron irradiation.

The U.S. Geological Survey has made analyses of many streams in the United States. Analyses of some of the larger streams have been presented by Love and White (Love, S. K., and W. F. White, *Analyses of Some Larger Streams of the United States at High and Low Flows*, *Ind. Eng. Chem.*, vol. 48, p. 1248, 1956) and are shown in Table 21-2.

Selection of the site must be based on the optimum physical and chemical characteristics of water in both quality and quantity. Hydrologic investigations will lead to a knowledge of maximum and minimum flows. These are necessary in terms of adequacy of supply for cooling water and other industrial purposes. They are also needed for dilution of the radioactive wastes which may be discharged to the stream.

Natural purification processes in surface waters such as lakes and streams will cause some removal of radioactivity from the water. The self-purification phenomena may include sedimentation, adsorption on the bottom muds or vegetation, and ingestion by aquatic organisms. Dispersion by mixing and dilution may be sufficient at an ideal site for low-level-waste discharge to assure radioactivity levels below tolerance values specified in the Federal Register, particularly at downstream points where time in the stream has been sufficient to bring about radioactive decay.

Thomas (Thomas, H. A., Jr., *Disposal of Low-level Liquid Radioactive Wastes in Inland Waterways*, AEC Rept. TID-7517, p. 457) has enumerated the factors which must be assessed in evaluating potential feasibility of discharging low-level liquid wastes to a stream. In selecting a plant site the following factors must be considered:

1. Hydrometeorological
 - a. Mean flow and annual variation
 - b. Mean temperature and annual variation; temperature stratification
 - c. Droughts and floods
 - d. Flow times; mixing characteristics
(longitudinal, lateral, and vertical dispersion)
2. Chemical, biochemical, and physical
 - a. Mineral substrate of stream and other factors relevant to the biological environment
 - b. Chemical composition of the waste (organic; inorganic)
 - c. Physical characteristics of the waste (dissolved, suspended, settling properties, temperature)
 - d. Rate of application of waste (intermittent; continuous, seasonal variation)
3. Radioactive characteristics (maximum permissible concentrations for ingestion and immersion)

4. Stream use; water treatment prior to use of stream water
5. Cost of disposal relative to alternate methods
6. Feasibility of procedure for monitoring and control (sampling of influent waste, bottom deposits, raw and treated water)

Ground waters must also be investigated at any potential site to determine the characteristics of the water-bearing formation. It is important to know the depth to ground water, the characteristic of the soils, and the rate at which water is moving through the aquifer. In normal water-bearing materials the natural gradients almost always are found to range from 0.001 to 0.01, with the most probable value closer to 0.001 (Theis, C. V., *Geologic and Hydrologic Factors in Ground Disposal of Waste, AEC Rept. Wash-275*). This represents a fall in the ground-water table of 5 ft/mile. Experience has shown that a good sandy aquifer will transmit water under natural conditions at a rate close to 1 gal/day/sq ft.

Actual velocities depend on soil porosities. For instance, with a 20 per cent porosity the average speed of water in a good aquifer is about $\frac{2}{3}$ ft/day. Brookhaven studies showed that the natural velocities in the glacial deposits on Long Island approximate $\frac{1}{2}$ ft/day. At Hanford, even with higher gradients, ground-water velocities were about the same order of magnitude (Theis, C. V., *Geologic and Hydrologic Factors in Ground Water Disposal of Waste, AEC Rept. Wash-275*). These velocities are very important where it is planned that surface disposal of radioactive wastes will be practiced and where mixing with ground water will occur.

Study of ground-water characteristics implies a geological survey of each potential site. This is also necessary for other purposes. **Geologic factors** are always considered in the construction foundations for nuclear plants in order to determine the bearing capacities of soils and other formations. The corrosivity of soils must be taken into account because underground storage tanks are generally included for the retention of liquid wastes at most nuclear-energy plants. Since retention may be for a long period of time, the foundation characteristics also exert considerable influence. The capacity of soils to absorb large volumes of low-level radioactive wastes from surface pits or cribs may also enter into the design. This is particularly true where the depth to the water table is quite great, as at Hanford, Wash. Permeability and exchange capacity of the soils underlying the nuclear plant must be coordinated with the rate and direction of flow of the ground waters discussed above. The possibility that these ground waters enter surface streams must also be known from geologic investigations. Experiments must be conducted to determine the relative rates of migration of various isotopes and chemical compounds emanating from the plants in order to determine the limits of liquid disposal of low-level wastes. These limits will be influenced by the proximity of public water supplies and the rate of decay of the radionuclides entering the ground-water stream.

Topography of the proposed site is important not only from the standpoint of the construction engineer but also for the potential ability of disposing of gaseous wastes. A flat open plain on relatively high ground may lead to more consistent atmospheric dilution of stack effluents and also be better for the disposal of liquid wastes to ground storage pits or underground strata. Deep valleys, even though they may contain large quantities of good water supplies, may lead to higher construction costs and more difficulty in the disposal of gaseous and liquid effluents.

Meteorological factors may also play an important role in site selection. The capacity of the atmosphere to dilute the various gases, aerosols, and particulate matter that may be released from fuel-fabrication or processing plants, research laboratories, and nuclear reactors must be studied by meteorologists for each type of plant and site location. Since radioactive wastes may be several orders of magnitude more toxic than the usual industrial stack discharges, it is necessary to determine in a quantitative manner the changes in air and ground concentrations over the wide range of atmospheric-dilution capacity which will be encountered at any site. The height and location of stacks play an important part in the over-all economy of the plant and thus influence site selection. In areas where temperature inversions are frequent and prolonged, as in river valleys, high stacks may be necessary to discharge the effluents above the inversion layer or in more turbulent air where diffusion is not limited.

The requirement for precise and detailed meteorological information is directly proportional to the magnitude of the pollution emanating from the proposed stacks, the size of the site, and the topography of the general area. For a small laboratory handling low-level radioactive materials, meteorology might not be an important factor.

Greater flexibility in site selection may be obtained if the proposed plant can be operated in such a manner that the gaseous effluents discharged can be controlled when meteorological conditions are not favorable. This may mean stopping the operations, or at least slowing them down in the event that forecasts would show above-tolerance ground concentrations of radioactivity if the plant were to be operating at full capacity. At Brookhaven National Laboratory and at Hanford routine forecasting has been adopted for the control of radioactive effluents. In most instances it is better to utilize efficient air-cleaning devices, high stacks, or isolated sites rather than depending upon favorable meteorological conditions for normal operations. In considering any proposed atomic-energy plant site, micro-meteorology may be more important than macrometeorology. The first deals with actual conditions such as the space-time variation of wind and temperature within a thousand feet of the surface of the earth in order to obtain qualitative and quantitative estimates of atmospheric diffusion. The latter term implies the gross weather conditions affecting the region in which the plant is to be located. Both are important and call for the services of experienced meteorologists on the team of engineers and scientists engaged in selecting the site.

LIQUID WASTES

Low-level wastes, for the purpose of this section, have been arbitrarily defined as all wastes other than those resulting from fuel-element processing and metal-recovery operations. The range of concentration of radioisotopes may thus vary from trace amounts of microcuries per liter to several millicuries per liter. In certain instances, wastes falling under this definition of low-level wastes will need to be handled as high-level wastes. Similarly, some incidental wastes associated with fuel processing fall in the category of low-level wastes.

Wastes which may be characterized as low-level include those resulting from ore processing, decontamination work (laundering of clothing, cell decontamination

work, etc.), pile operations (pile coolant and canal water), laboratory research, diagnostic and therapeutic work in hospitals and physicians' offices, and certain phases of fuel-decontamination operations. The problems associated with treatment of these wastes arise for the most part because of small concentrations of radioactivity in large total volumes of water, or because of difficulties in control and collection, and not because of the total curie content of the waste. The diversity of the sources from which low-level wastes arise makes them extremely variable in character. The general characteristics of a few types of low-level wastes are given below.

Ore processing results in considerable volumes of low-specific-activity wastes containing the alpha-emitting isotopes of uranium. A by-product of the decay of the associated radium is the gaseous product radon, which can be a health hazard after prolonged exposure. At present these wastes are stored in concrete basins.

Decontamination wastes are usually characterized by large volume and small quantity of radioactivity, the result of flushing or rinsing to effect the decontamination. The initial wash will contain a high concentration of the decontaminating agent, usually acid or one of a number of complexing agents, and the bulk of the active material. If the contaminated object was previously sandblasted or brushed, solid material may also be found in the waste rinse water.

Cooling fluids for operating nuclear reactors become radioactive largely as the result of neutron bombardment of the coolant itself. If the coolant is water, inert salts present will become radioactive. In addition, corrosion of materials in the reactor adjacent to the coolant flow will release radioactive particulate matter into the cooling medium. Radioactive fission products may gain access to the coolant in the event of a slug rupture. Essentially the same types of contamination, probably in lesser amounts, may be expected in the water of the canal used to catch and shield spent fuel elements and target materials.

Laboratory wastes will contain all the chemical compounds incidental to the operation of a chemical, metallurgical, or biological laboratory in addition to greater or lesser amounts of radioactivity. The waste may also contain relatively large proportions of stable isotopes used either to precipitate active material or as a hold-back carrier to prevent coprecipitation of unwanted activity. As a general rule, laboratory wastes with high levels of activity will be small in volume. Decontamination of glassware, gloves, and equipment, on the other hand, will produce large volumes of extremely low level waste.

The uses of **radioisotopes in hospitals** generally fall into one of two categories—diagnostic and therapeutic. In diagnostic work, the amount of active material used is small so that no active waste of importance results from this application. In therapeutic work other than that performed with solid encapsulated or sealed sources (i.e., Co-60 or radium) rather large dosages of radioisotopes are administered internally. Depending on the initial retention of the material by the patient, and on the natural and biological half-lives of the isotope used, the excreta of the patient may present a problem in disposal, although the AEC exempts them from specific control. The isotopes used most frequently in therapeutic work are Iodine-131 and Phosphorus-32, both of which have relatively short half-lives. Other isotopes, which are used in considerably smaller quantities, are Gold-198, which presents no

problem because of its short half-life and complete retention by the body; and Chromium-51, with a half-life of about 27 days.

In **fuel-element processing**, the major part of the fission products appear in the first decontamination step. Subsequent operations may produce wastes with low levels of radioactivity, which may or may not be combined with the highly active

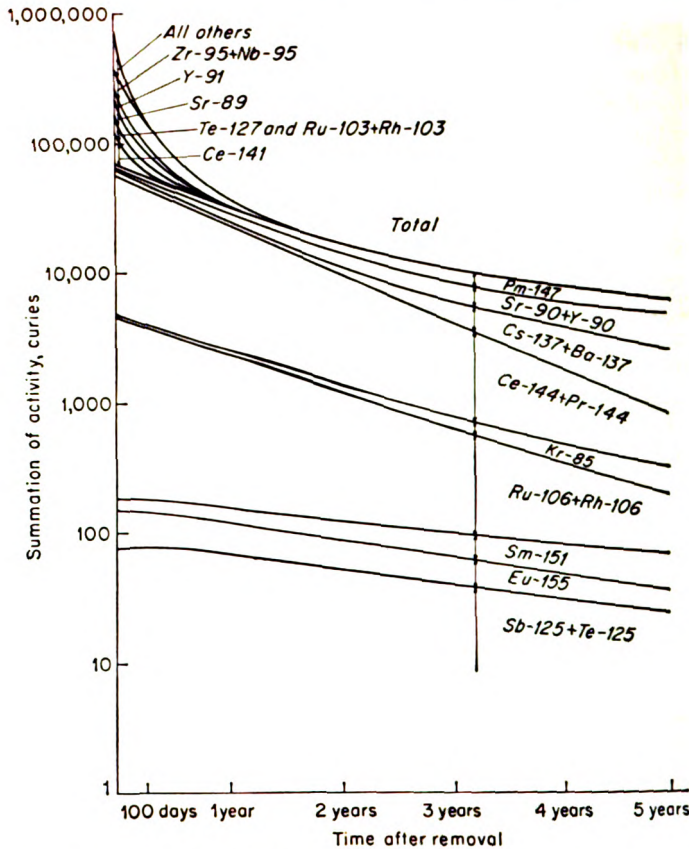


FIG. 21-1. Activity-time relationship of specific fission products after removal from a reactor which operated at 1,000 kw for 1 year. [Terrill, J. G., D. W. Moeller, and S. C. Ingraham, *Fission Products from Nuclear Reactors (A Quantitative Prospectus)*, Proc. Am. Soc. Civil Engrs. Separate 643, p. 38, March, 1955.]

first-stage waste. Scrubbing of off gases to remove volatile fission products and entrained spray, and solvent cleaning operations produce additional low-level wastes.

High-level liquid wastes result from the processing of spent fuel elements to recover fertile and fissionable material. Processing of reactor fuel is necessary because of damage to reactor materials through corrosion, radiation, or heat, and because of the poisoning effects of neutron-absorbing fission products which accumu-

late in the reactor. Normally, spent fuel material is allowed to cool for a considerable period to permit short-lived radioisotopes to decay.

As shown in Fig. 21-1, the rate of decay is very rapid immediately following removal of the fuel from the reactor, but decreases as the composite effect of the longer-lived isotopes becomes predominant. [Terrill, J. G., D. W. Moeller, and S. C. Ingraham, *Fission Products from Nuclear Reactors (A Quantitative Prospectus)*, *Proc. Am. Soc. Civil Engrs.*, Separate 643, p. 38, March, 1955.]

Fuel Processing. The process used to decontaminate and recover spent fuel material depends on the type of reactor from which the material was taken. Most of the present reactors are of the heterogeneous type, with the solid fissionable material either clad with a metal jacket (such as Al) or in the form of an alloy. In some instances, with clad fuel elements, the jacket can be stripped either mechanically or chemically from the fuel rod and the fuel dissolved separately. With alloy-type fuel elements, the entire slug is dissolved in acid solution in one step. Solids in the acid solution are removed by centrifuging or filtering, producing a small amount of waste.

After dissolution of the fuel element the usual method of metal recovery and decontamination is by selective extraction of highly oxidized uranium or plutonium by solvents such as diethyl ether or tributyl phosphate. From nitric acid solutions, uranium and plutonium are extracted as nitrate complexes. In order for the extraction to be quantitative it is necessary to add inert nitrates to the aqueous phase. The "salting" agent used may be nitric acid alone or aluminum nitrate. With acid alone and with clad elements, the waste stream can be made essentially solids-free (except for fission products) by stripping the jacket from the fuel assembly before dissolution. If aluminum nitrate is used as the salting agent, the fuel is dissolved without removing the cladding. The organic phase leaving the extraction column contains essentially all the fissionable material, while the fission products, acid and bulk salts, such as aluminum, remain in the aqueous phase. The fuel-bearing organic stream is further decontaminated by scrubbing with a nitrate-bearing aqueous phase. Several extraction cycles may be necessary to reduce the fission-product contamination in the uranium to acceptable levels. The first-extraction-cycle raffinate will contain more than 99.9 per cent of the total fission-product activity. The aqueous streams from subsequent extraction steps will contain most of the remaining 0.1 per cent. The strength and volume of the high-level waste produced will depend on the number of extractions and on whether or not the various waste streams are combined.

In some **plutonium-recovery** operations, uranium remains in the waste stream with the fission products. This waste is usually neutralized with sodium carbonate and sodium hydroxide and stored, during which time precipitation of uranium and a number of the fission products occurs. The uranium can be recovered by solvent extraction after the concentrated sludge is dissolved in nitric acid. The waste stream from this system contains, in addition to aged fission products, nitric acid and sodium nitrate.

Chemical and physical characteristics of typical high-level wastes are given in Tables 21-3 to 21-5 (Culler, F. L., *Notes on Fission Product Wastes from Proposed Power Reactors, Oak Ridge Natl. Lab. Rept. CF 55-4-25*, 1955).

Table 21-3. Physical Nature of High-level Radioactive Wastes *

Properties	Natural U-Pu		Enriched U-235 diluent salted
	Salted	HNO ₃ salted	
Boiling point, °C.....	102	112	103
Specific gravity.....	1.18	1.24	1.23
Concentration possible.....	3	?	2
Gal/g U-235 burned (unneutralized)...	2.5	0.5	5.0
Gal/g U-235 burned (neutralized with NaOH solution).....	10.0	1.5-2.0	20.0

* Culler, F. L., Notes on Fission Product Wastes from Proposed Power Reactors, Oak Ridge Natl. Lab. Rept. CF 55-4-25, 1955.

Table 21-4. Chemical Nature of High-level Radioactive Wastes *

Properties	Natural U-Pu		Enriched U-235 diluent salted
	Salted	HNO ₃ salted	
Acidity <i>N</i>	-0.3 (basic)	8.0	-0.2
Total salts <i>M</i> (unneutralized).....	1.2	0.2	2.0
Total salts <i>M</i> (neutralized).....	6.0	8.2	Acid storage
Solids stability.....	Unstable basic	Stable	Unstable basic

* Culler, F. L., Notes on Fission Product Wastes from Proposed Power Reactors, Oak Ridge Natl. Lab. Rept. CF 55-4-25, 1955.

Table 21-5. Radiochemical Nature of High-level Radioactive Wastes *

Properties	Natural U-Pu		Enriched U-235 diluent salted
	Salted	HNO ₃ salted	
Curies/gal acid.....	80	400	2,000
Curies/gal basic.....	20	200	Acid storage
Shielding, lead, 1 cc acid, in.....	3.5	4	4
Shielding, lead, 500 gal acid, in.....	11.5	12	12
Watts/gal acid.....	0.3	1.2	5.4

* Culler, F. L., Notes on Fission Product Wastes from Proposed Power Reactors, Oak Ridge Natl. Lab. Rept. CF 55-4-25, 1955.

CHEMICAL PRECIPITATION OF RADIOACTIVE WASTES

Chemical-precipitation techniques for fission-product waste concentration fall into two categories:

1. Direct precipitation of specific isotopes
2. Concentration by adsorption or mixed-crystal formation using a carrier precipitate

Recovery of specific fission products in large quantities by direct precipitation appears to be justified economically for only a few of the long-lived isotopes. Processes for the recovery of strontium and cesium have received considerable attention for two reasons: (1) these isotopes may be of value as radiation sources; and (2) their separation reduces the hazards of disposal of the remainder of the waste.

Strontium may be precipitated from concentrated nitric acid solution (Glueckauf, E., and T. V. Healy, "Chemical Processing of Fission Product Solutions," Paper

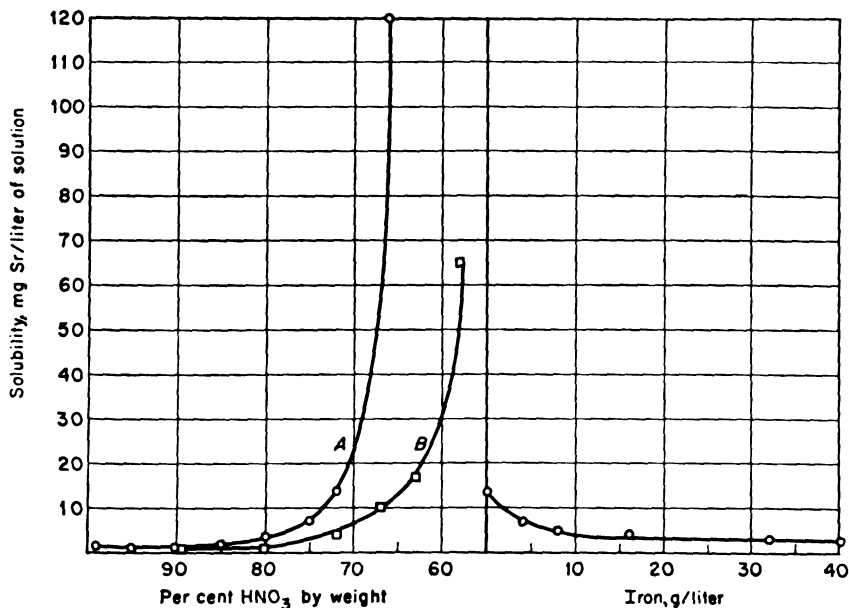


FIG. 21-2a. Solubility of strontium as a function of HNO₃ concentration. (A) In HNO₃. (B) With 16g/liter iron. (Glueckauf, E., and T. V. Healy, "Chemical Processing of Fission Product Solutions," *Proc. Intern. Conf. Peaceful Uses Atomic Energy*, Geneva, Paper P/415, vol. 9, pp. 635-639, 1956.)

FIG. 21-2b. Solubility of Sr in 72 per cent HNO₃ as a function of iron concentration. (Glueckauf, E., and T. V. Healy, "Chemical Processing of Fission Product Solutions," *Proc. Intern. Conf. Peaceful Uses Atomic Energy*, Geneva, Paper P/415, vol. 9, pp. 635-639, 1956.)

P/415, vol. 9, pp. 635-639, International Conference on Peaceful Uses of Atomic Energy, Geneva, 1956), directly as the nitrate, either alone or with the addition of inert carrier or a salting agent such as iron. The solubility of strontium in HNO₃

and the influence of added iron are shown in Fig. 21-2. Radioactive strontium can be precipitated to an even greater extent by the addition of carrier Sr, Ba, or Pb, with a corresponding decrease in the specific activity of the precipitate.

Cesium may be precipitated from strong nitric acid solutions of the fission products as cesium phosphotungstate with yields from 8*N* HNO₃ of 99.5 per cent. Decontamination from other fission products is 99.9 per cent. Cesium may also be recovered by precipitating cesium alum in the presence of high concentrations of the lighter alkali alums. Ninety to ninety-five per cent of the cesium is precipitated on cooling a warm solution of potash alum to 25°C. (Gresky, A., Recovery of Cs¹³⁷ from ORNL Radiochemical Waste, *Oak Ridge Natl. Lab. Rept. 742* or *AEC Rept. AECD-2899*.)

Concentration of the radioactivity of wastes by precipitation of a **scavenger precipitate** has been successful for **high-volume low-level wastes** where direct precipitation is not feasible because of the small quantities of radioactive material present. The processes by which carrying of foreign ions by a precipitate occur have been classified (Hahn, O., "Applied Radioactivity," Cornell University Press, Ithaca, N.Y., 1936) as follows:

(a) Isomorphous replacement, in which the foreign ion is incorporated into the crystal lattice of the precipitate.

(b) Adsorption, in which the trace element is attracted to the precipitate by surface forces.

(c) Internal adsorption, in which the foreign ion is adsorbed on the surface of the precipitate as it is in the process of growing. The tracer is thus incorporated into the precipitate, but not as in (a).

(d) Anomalous mixed crystal formation, in which the trace material is apparently incorporated into the crystal as in (a), although such incorporation is not observed when macro amounts of the trace element and carrier are precipitated together.

Isomorphous replacement may be expected to occur in cases where the trace ion and carrier ion are not only similar in size and charge, but also crystallize in the same form with nearly identical angles. For example, strontium will be precipitated more completely with calcium as the carbonate if the calcium is in the form of aragonite, rather than calcite, since both aragonite and strontium carbonate (strontianite) form rhombic crystals. In cases of coprecipitation by isomorphous replacement, the method of precipitation is important because of the negligible rates of diffusion within the crystal. Hence, equilibrium conditions should be maintained at all times during the course of the precipitation.

Surface adsorption of trace ions by a scavenger precipitate is affected by a number of factors, as follows:

1. In general, the more insoluble the compound formed by the trace ion with the ion of the opposite charge in the lattice of the precipitate, the greater will be the efficiency of carrying.

2. The larger the surface area of the precipitate the better the foreign ion will be carried.

3. For efficient carrying the precipitate must carry a charge opposite to that of the ion to be removed.

4. The higher the charge on the trace ion, the greater will be the fraction removed from solution, assuming that condition 3 has been fulfilled.

5. Prolonged standing or digestion may either increase or decrease the amount of trace ion carried but will usually result in a decrease due to recrystallization and growth of the small crystals, which results in a smaller total surface area.

Internal adsorption and anomalous mixed-crystal formation are special cases of (a) and (b) and need no further discussion.

As a very general rule, it may be expected that trace elements will be carried or coprecipitated by a scavenger precipitate if the conditions of precipitation are such that the tracer would precipitate if present in macroamounts.

Coprecipitation techniques to concentrate radioactive materials have been used in only a few instances in waste-treatment facilities for two reasons:

1. Such techniques are of most value in the treatment of large-volume wastes, most of which should present no major problems of disposal at properly selected sites.

2. Because of the widely varying characteristics of the chemical species present, most scavenging procedures are relatively inefficient for gross removal of fission products or fissionable material.

At the Los Alamos National Laboratory a waste-treatment facility was constructed to handle **wastes containing plutonium** in concentrations varying from

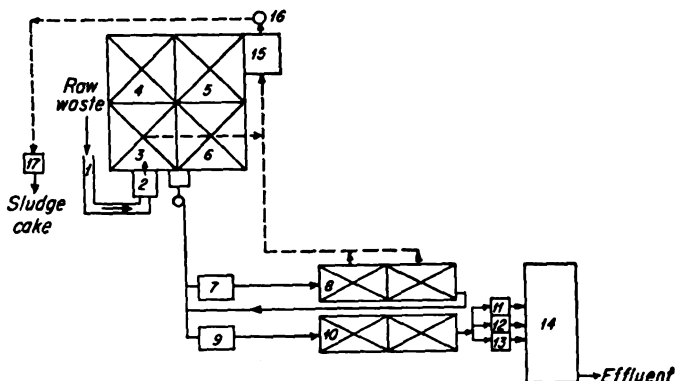


FIG. 21-3. Flow diagram of waste-treatment plant—Los Alamos Scientific Laboratory. (1) Influent weir channel. (2) Flash mixer. (3, 4, 5, 6) Influent-holding tanks. (7) Primary flocculator. (8) Primary sedimentation basin. (9) Secondary flocculator. (10) Secondary sedimentation basin. (11, 12, 13) Filters. (14) Effluent-holding tanks. (15) Sludge basin. (16) Diaphragm sludge pump. (17) Vacuum filter. (*AEC Rept. Wash-275, p. 201, 1954.*)

200 to 1,000,000 disintegrations/min/liter of waste (Wash-275, Sanitary Engineering Conference, Baltimore, Md., Apr. 15-16, 1954, Office of Technical Services, Department of Commerce). The maximum permissible concentration of plutonium in the effluent established at Los Alamos is 330 disintegrations/min/liter, equivalent to 2.4×10^{-3} $\mu\text{g/liter}$. Because of a diversity of operations in the laboratory the pH of the waste may vary from 2 to 13, and the total solids content from 230 to 8,000 ppm. The flow diagram of the plant, for treating an average of 30,000 gal/day, is shown in Fig. 21-3. The waste flows into the plant over a weir (1) where pH and flow measurements are made and recorded. The waste then moves to a

mixer (2) in which lime and/or caustic soda are added to adjust the pH. Tanks (3), (4), (5), (6) are holding tanks to equalize the flow and have a capacity equal to 2 days' flow. One of the tanks can be used to hold unusual wastes which are difficult to treat so that they may be fed gradually into the main waste stream. Solids which settle and accumulate in the bottom of the holding tanks can be withdrawn through hopper bottoms. The neutralized waste is pumped to the flocculators (7) and (9), where chemicals are added to produce a flocculent precipitate of large surface area. The flocculators and the sedimentation basins (8) and (10) may be operated either in series or in parallel. The theoretical detention time of the plant during series operation at a flow of 35 gal/min is 30 min in each flocculator and 6 hr in each sedimentation basin. The settled effluent flows to three rapid sand filters (11), (12), and (13), each 17 sq ft in area. Two of the filters use sand (effective size = 0.48 mm; uniformity coefficient ≤ 1.7); the other contains Anthra-filt (effective size = 0.60; uniformity coefficient ≤ 1.75). The filters are equipped with loss-of-head gauges, flow controllers, backwashing facilities, and a surface-wash system. The rate of filtration has been varied from 1 to 3 gal/min/sq ft of sand surface. The filter effluent is collected in holding tanks (14) to be monitored for residual activity before discharge into a canyon. Effluent water used to backwash the filters is returned to the influent holding tanks for retreatment. Sludge which has settled out in the sedimentation basins flows to a sludge-concentration tank where it is allowed to compact further by sedimentation. The concentrated sludge is further dewatered by vacuum filtration, and the liquid is returned for retreatment. Typical operation costs for the plant are shown in Table 21-6. The

Table 21-6. Typical Annual Operating Costs, Los Alamos Waste Treatment Plant (WD-2) *

Total days operated.....	262
Chemicals.....	\$ 1,503
Labor.....	5,150
Supervision.....	2,400
Laboratory analyses.....	4,200
Depreciation.....	20,004
Maintenance.....	4,200
Utilities.....	1,280
Sludge burial.....	793
Miscellaneous.....	835
Total.....	\$40,365
Cost/1,000 gal.....	\$ 5.86

* Wash-275, p. 204, Sanitary Engineering Conference, Baltimore, Md., Apr. 15-16, 1954, Office of Technical Services, Department of Commerce.

initial cost of the plant was \$200,000, with an additional \$150,000 for laboratory facilities. The **operating costs** shown are considerably higher than for municipal plants using similar treatment methods. This is due not only to the close supervision required and the high capital cost per 1,000 gal of water treated but also to the fact that the plant was designed and has been used for experimental research.

During a typical operating period the average flow to the plant was 70 gal/min containing an average of $\sim 8,500$ disintegrations/min/liter of plutonium. The average plutonium content of the effluent was 98 disintegrations/min/liter, or approximately one-third the tolerance value. The sludge produced from treating a total volume of approximately 7,000,000 gal of waste was 1,900 cu ft.

The chief problem in the operation of the plant arises from the presence of complexing agents such as citric acid in the waste. These inhibit the carrying of Pu by the floc. The problem has been overcome by precipitating at a pH of about 11

Table 21-7. Removal of Radioactivity, Chemical Coagulation *
(Ferrous sulfate, ferric chloride, and aluminum sulfate at 1, 2, and 6 gpg)†

Isotope	No. of tests	Activity range, curie/min/ml	pH range		% removal	
			Initial	Final	Avg	Range
Cs ¹³⁷ -Ba ¹³⁷ (Cl)	27	19-17,390	7.5-7.8	6.5- 7.8	0.5	0-37
Sr ⁸⁹ (Cl ₂)	54	212-25,950	7.7-7.9	7.5- 8.0	3	0-15
Ba ¹⁴⁰ -La ¹⁴⁰ (Cl ₂)	81	42-17,150	7.4-7.6	7.2- 8.2	59	1-84
	54	42- 5,520	7.4-7.6	6.6- 7.9	44	19-59
	54	44- 8,070	7.4-7.6	7.2- 7.8	52	27-72
Sc ⁴⁶ (Cl ₃)	27	163-24,300	7.2-7.8	6.4- 8.0	89	62-98
	27	326-20,060	7.6	7.0- 7.8	92	62-99+
	27	202-26,100	7.5-7.9	6.9- 8.2	91	69-99+
Y ⁹¹ (Cl ₂)	54‡	188- 4,166	7.5-7.6	6.0- 7.6	90	40-99
	81‡	367-20,860	8.1	7.8-10.2	92	56-99+
	81‡	202-21,450	7.8-8.0	7.6- 8.4	86	48-99
	81¶	406-21,500	7.8	7.5- 9.1	75	1-98
	81¶	222-21,500	7.1-8.4	7.4- 9.5	73	33-98
Zr ⁹⁵ -Nb ⁹⁵ (oxylate complex)	54	186-27,200	7.4-7.6	6.4- 7.9	80	2-99
P ³² as PO ₄	27	228-17,625	7.6	7.4- 7.8	96	68-99+
Cr ⁵¹ (Cl ₂)	54	5- 758	6.7-7.3	5.3- 7.8	6	0-60
Mo ⁹⁰ (O ₃)	54	84-10,085	7.6	7.0- 8.2	10	0-60
W ¹⁸⁵ (tungstate)	54	233-28,290	7.5-7.9	6.6- 8.4	46	1-96
	81	198-22,490	8.0-8.1	7.5- 8.4	36	4-91
Re ¹⁸⁶ (metal)	54	95-15,230	7.6	7.2- 8.0	6	0-29
I ¹³¹ (iodide)	27	239-25,060	7.6	7.2- 7.8	20	0-44
Ru ¹⁰³ (Cl ₃)	27	196-24,960	7.3-7.6	6.9- 8.2	77	43-96
Pr ¹⁴² (as Pr ₂ O ₃)	27	102-12,050	7.3	7.4- 8.0	96	83-99+
Ce ¹⁴⁴ -Pr ¹⁴⁴ (Cl ₃)	27	244-30,510	7.4-7.6	7.2- 8.4	91	28-99+
Pm ¹⁴⁷ (Cl ₃)	27	136-19,040	7.4-7.6	7.0- 8.2	87	4-99+
Sm ¹⁵³ (as Sm ₂ O ₃)	54	148-20,870	7.4-7.6	5.8- 8.0	91	44-99+
F.P. mixture	27	216-32,320	7.4-7.6	7.4- 8.2	73	46-86

* Morton, R. J., and C. P. Straub, *J. Am. Water Works Assoc.*, vol. 48, no. 5, May, 1956, by permission of AWWA.

† 1 gpg means 1 gpg of either FeSO₄, FeCl₃, or alum +0.4 gpg activated sodium silicate +1 gpg of Ca(OH)₂ or Na₂CO₃. 2 gpg and 6 gpg are multiples of these amounts.

‡ Ca(OH)₂ was used as alkaline agent.

¶ Na₂CO₃ was used as alkaline agent.

with larger dosages of coagulant and lime. Under these conditions severe incrustation of the sand in the filters occurred (the Anthrafil to a lesser extent). This was controlled by the addition of Calgon to the waste to sequester the incrusting substances.

Research on the efficacy of the conventional water-treatment process and others has led to the accumulation of a large amount of data (Morton, R. J., and C. P. Straub, Removal of Radionuclides from Water by Water Treatment Processes, *J. Am. Water Works Assoc.*, vol. 48, no. 5, p. 545, 1956) on the removal of specific isotopes, as well as mixed fission products. In Tables 21-7 to 21-11, reference to coagulations by additions of ferric chloride or ferrous sulfate indicate the precipitation of

Table 21-8. Coagulation and Settling Results, Jar-test Studies *

Isotope	Clay added, ppm †	Coagulant added, gpg ‡	Final pH	% removal
Cs ¹³⁷ -Ba ¹³⁷	0	1		0-6
	100	1		35-65
Sr ⁸⁹	0	1.5¶	6.7-7.8	0-6
	100	0.5-6	6.7-10.7	0-51
Cd ¹¹⁵	0	1		40-60
	100	1-5		60-95
Ba ¹⁴⁰ -La ¹⁴⁰	100	1-6	7.5-8.2	28-84
	100	1-6	6.5-8.2	66-98
Sc ⁴⁶	0	1.5¶	6.8-7.1	83-93
	100	1-6	7.0-10.2	34-99
Zr ⁹⁵ -Nb ⁹⁵	0	1-5		70-98
	100	1		95-99
P ³²	100	0.5-1.5¶	6.8-8.8	97-99
Cr ⁵¹	100	1-6	7.6-8.8	73-98
W ¹⁸⁵	100	1-6	7.5-8.4	5-91
I ¹³¹	100	0.5-2¶	6.9-9.0	0-10
Ce ¹⁴⁴	0	1-1.5¶	7.2-7.8	81-94
	100	0.5-2.5¶	7.0-7.8	85-96
FPM-1	100	1-3¶	7.2-8.8	61-84
FPM-2	0	1-5	4.3-10.2	9-71
	100	1-5	4.3-10.2	12-73
FPM-3	0	10	9.9-10.0	46

* Morton, R. J., and C. P. Straub, *J. Am. Water Works Assoc.*, vol. 48, no. 5, May, 1956, by permission of AWWA.

† Local clay was added.

‡ Coagulant includes alum, ferrous sulfate, or ferric chloride, lime, soda ash, or sodium hydroxide, and sodium silicate.

¶ No sodium silicate added. Where added sodium silicate equals 40% of primary coagulant dose.

FPM-1: iodine dissolver solution.

FPM-2: synthetic mixture containing fission products in the same concentrations assumed to be present 30 days after an underwater bomb blast.

FPM-3: Three-year-old fission-product mixture.

ferric hydroxide, or a hydrous ferric oxide; aluminum sulfate addition leads to precipitation of aluminum hydroxide, or a hydrous oxide of aluminum. Turbidity refers to the amount of clay material added to improve floc formation or to increase the removal of specific isotopes through adsorption or ion exchange. In the phosphate coagulation experiments, a calcium phosphate precipitate was obtained by adding sodium phosphate to a water containing lime at high pH. An excess of phosphate was desirable for greatest efficiency.

Table 21-9. Strontium Removal by Chemical Treatments *

Type of Treatment	Sr removal, %		No. of cities
	Av.	Range	
Alum or ferrous sulfate	12	10-31	7
Alum or ferrous sulfate, plus lime	37	10-75	11
Alum or ferrous sulfate, plus lime and soda ash	54	10-85	3
Alum or ferrous sulfate, plus lime and phosphate	42	10-70	5
Softening only (phosphate, ion exchanger)	73	69-76	2
None (except chlorine, fluoride, carbon, or ammonia)	10		8

* Morton, R. J., and C. P. Straub, *J. Am. Water Works Assoc.*, vol. 48, no. 5, May, 1956, by permission of AWWA.

Table 21-10. Removal of Radioisotopes by Phosphate Coagulation *

Isotope	Coagulant	Coagulant dose, mg/liter	Removal, %
Ce ¹⁴⁴	KH ₂ PO ₄	200	99.8
	Na ₃ PO ₄	120	99.9
Sr ⁸⁹	KH ₂ PO ₄	100	81.3
	Na ₃ PO ₄	240	97.8
Y ⁹¹	KH ₂ PO ₄	100	99.9
Sb ¹²⁴	KH ₂ PO ₄	100	66.1
	Na ₃ PO ₄	120	67.4
Zn ⁶⁵	KH ₂ PO ₄	50	99.6
W ¹⁸⁵	KH ₂ PO ₄	200	10.7
Zr ⁹⁵	KH ₂ PO ₄	100	99.5
Nb ⁹⁵	KH ₂ PO ₄	100	99.2

* Morton, R. J., and C. P. Straub, *J. Am. Water Works Assoc.*, vol. 48, no. 5, May, 1956, by permission of AWWA.

Table 21-11. Laboratory Studies on Removal of Radioactivity from Oak Ridge National Laboratory Process Wastes *

Type of sample	No. of observations	Treatment	% removal	
			Avg	Range
Grab †.....	5	Alum coagulation	28	23-32
Grab †.....	5	Ferrous sulfate coagulation	32	15-44
Grab †.....	5	Phosphate coagulation	73	66-82
Grab †.....	4	Excess lime-soda ash	70	61-78
Grab †.....	1	Variable clay (100-1,000 ppm)	..	39-52
Grab.....	20	Phosphate coagulation	70	43-83
8-hr composite..	6	Phosphate coagulation	70	46-78
8-hr composite..	6	Excess lime-soda ash	37	19-48
8-hr composite..	7	Serial treatment—clay + phosphate coagulation	90	82-93
8-hr composite..	6	Serial treatment—clay + excess lime-soda ash softening	84	82-88

* Morton, R. J., and C. P. Straub, *J. Am. Water Works Assoc.*, vol. 48, no. 5, May, 1956, by permission of AWWA.

† Preliminary tests.

ION EXCHANGE

The use of ion-exchange resins for the treatment of radioactive wastes has many advantages in theory. Among these are large volume reduction, substantial decontamination factors, and ease of operation. In actual practice the use of exchangers has been severely limited for a number of reasons. In some cases the waste contains high concentrations of inert salts which exhaust the resin or decrease the removal of monovalent or divalent radioisotopes. In others, it has been found necessary to pretreat the waste to remove suspended and colloidal solids. Often it has been found that the resins are difficult to regenerate without the use of large volumes of strong acids. In the treatment of high-level wastes the use of ion-exchange resins has been limited by direct-radiation effects to the resin, which result in a decrease in exchange capacity and degradation of the physical and chemical characteristics of the resin (Cathers, G. I., *Radiation Damage to Radiochemical Processing Reagents, Proc. Intern. Conf. Peaceful Uses Atomic Energy*, vol. 7, pp. 490-495, 1956).

For materials such as clays and zeolites the crystal-lattice theory and the concept of ionic solids have been used to explain many of the phenomena which have been observed. This concept visualizes that in an ionic solid the components of the crystal lattice are ionic and not molecular. Each ion is surrounded by an equivalent number of oppositely charged ions in order to maintain electrical neutrality. Ions at the surface of a crystal are subject to fewer attractive forces than ions within the crystal. In polar media these forces are still further reduced. It is because of these factors that exchange of surface ions for ions in solution becomes possible. Factors which determine the relative ease with which one ion exchanges

for another are (1) the nature of the forces holding the exchangeable ion in the crystal lattice, (2) the chemical nature of the exchanging ion, and (3) the relative sizes of the ions involved. The exchange capacity per unit weight of a given mineral is dependent on the degree of subdivision of the mineral or, more precisely, on the accessibility of the exchangeable ions in the crystal.

The synthetic exchange materials are insoluble, highly polymerized, and cross-linked resins containing large numbers of diffusible cations, in the case of cation exchangers, or anions, in the case of anion exchangers. The high exchange capacity, physical stability, and ease of regeneration of exchangers led to their application to a wide variety of uses, both in waste treatment and in recovery of valuable materials.

The relative affinities of various ions involved in ion-exchange phenomena are similar to those found in the exchange in clay minerals, that is, $\text{Li} < \text{Na} < \text{K} < \text{Rb} < \text{Cs}$; $\text{Mg} < \text{Ca} < \text{Sr} < \text{Ba}$; $\text{Al} < \text{Sc} < \text{Y} < \text{Eu} < \text{Sm} < \text{Nd} < \text{Pr} < \text{Ce} < \text{La}$; $\text{F} < \text{Cl} < \text{Br} < \text{I}$. In concentrated solution, in nonaqueous media, or at high temperature, order of replacement is not necessarily as shown in the series given above.

Following is a qualitative set of empirical rules useful in predicting the performance of ion-exchange materials in dilute aqueous solution at ordinary temperatures (Kunin R., and R. J. Myers, "Ion Exchange Resins," chap. 2, p. 25, Wiley, New York, 1950).

1. The extent of exchange increases as the valence of the exchanging ion: $\text{Na}^+ < \text{Ca}^{++} < \text{Al}^{3+} < \text{Th}^{4+}$.

2. For ions of the same valence, the extent of exchange increases with increasing atomic number: $\text{Li} < \text{Na} < \text{K} < \text{Rb} < \text{Cs}$; $\text{Mg} < \text{Ca} < \text{Sr} < \text{Ba}$.

3. At high concentrations the order of exchange may be reversed; i.e., a monovalent ion may have a stronger exchange potential than a divalent ion.

4. At higher temperatures, differences in the exchange potential of ions of the same valence become smaller.

At the Argonne National Laboratory a 12-in.-diameter ion-exchange unit containing 3 cu ft of **Nalcite HCR** has been used for partial decontamination of a portion of the laboratory waste. (Rodger, W. A., *et al.*, Disposal of Radioactive Wastes at Argonne National Laboratory, *Proc. Eighth Ind. Waste Conf., Purdue Univ.*, p. 474, 1953. Swope, H. Gladys, and E. Anderson, Cation Exchange Removal of Radioactivity from Wastes, *Ind. Eng. Chem.*, vol. 47, no. 1, pp. 78-83, 1955. Swope, H. Gladys, Removal of Radioactivity by Ion Exchange, *Proc. Ninth Ind. Waste Conf., Purdue Univ.*, p. 118, 1954.) The drains from all laboratory sinks are connected to 1,500-gal **retention tanks**, which are monitored before discharge. Those with activity below tolerance are directed to the laboratory waste-treatment plant; the remainder are further treated. Wastes containing less than 3,000 ppm total solids and less than 1.5×10^{-4} $\mu\text{c}/\text{ml}$ gross beta-gamma radiation are filtered and then passed through the cation-exchange column. If the total beta count in the column effluent is less than 4×10^{-5} $\mu\text{c}/\text{ml}$, the waste is discharged to the laboratory waste-treatment plant; if greater than the acceptable limit, it is further treated. The results obtained by this treatment are given in Table 21-12 (Swope, H. Gladys, Removal of Radioactivity by Ion Exchange, *Proc. Ninth Ind. Waste*

Table 21-12. Ion-exchange Treatment of Radioactive Laboratory Wastes *

Filter: Alsop, 12 in. diameter, paper No. 40

Resin column: 12 in. diameter, containing 3 cu ft Nalcite HCR-Na⁺

Batch volume, gal	pH	Filtered feed, analysis			Effluent β activity, $\mu\text{c/ml}$	Removal, %
		Hardness as CaCO_3 , ppm	Total solids, ppm †	β activity, $\mu\text{c/ml}$		
920	2.3	0	2,420	2.4×10^{-4}	5.9×10^{-6}	98
1,000	2.5	22	1,660	1.6×10^{-4}	4.0×10^{-6}	75
1,310	2.7	26	2,120	9.9×10^{-5}	1.5×10^{-5}	85
1,000	3.2	111	1,100	1.4×10^{-4}	6.3×10^{-6}	54
800	4.4 †	21	880	1.6×10^{-4}	8.8×10^{-6}	40
1,320	3.1 †	102	1,000	8.2×10^{-5}	2.1×10^{-5}	74
1,540	7.4 †	102	860	2.4×10^{-4}	3.6×10^{-5}	85
1,200	2.3	144	1,000	2.1×10^{-4}	7.9×10^{-5}	62
1,120	1.9	111	3,680	2.5×10^{-4}	2.1×10^{-4}	17
920	2.1	150	1,610	1.2×10^{-4}	3.8×10^{-5}	69
1,180	2.3	141	1,070	9.8×10^{-5}	7.8×10^{-5}	21
1,420	2.8	131	1,125	9.3×10^{-5}	4.3×10^{-5}	54
1,080	1.9 †	¶	2,090	1.1×10^{-4}	2.6×10^{-5}	77
1,080	1.9 †	¶	1,860	9.4×10^{-5}	2.1×10^{-5}	78
1,700	3.2 †	148	1,020	2.4×10^{-4}	4.3×10^{-5}	82
1,150	3.6	195	1,240	3.6×10^{-4}	4.8×10^{-5}	87
$\Sigma 18,740$						
Avg	3.0	106	1,550	1.7×10^{-4}	4.8×10^{-5}	72

* Swope, H. Gladys, Removal of Radioactivity by Ion Exchange, *Proc. Ninth Ind. Waste Conf., Purdue Univ.*, p. 118, 1954.

† pH adjusted to ~ 2.5 before filtration.

‡ Planchet method $\pm 10\%$.

¶ Assumed hardness = 150.

Conf., Purdue Univ., p. 118, 1954). Use of this treatment for dilute wastes was based on laboratory results which indicated the following:

1. Flow rates as high as 10 gal/min/ft³ of resin were permissible.
2. Optimum pH for removal of gross fission products was 2.5.
3. At pH 2.5, no activity was removed by filtration of the feed. At pH 8, 41 per cent was removed by filtration; at pH 9.5, 67 per cent or more was removed.
4. Approximately 260,000 gal of waste/cu ft of resin could be processed with a beta decontamination of at least 76 per cent. This is due to continued removal of rare earths and most of the strontium after the column is exhausted with respect to calcium. Cesium is found in the effluent somewhat before the calcium breaks through.

A proposal for the treatment of **high-level radioactive wastes containing aluminum nitrate and nitric acid** has been demonstrated on a laboratory scale. (Higgins,

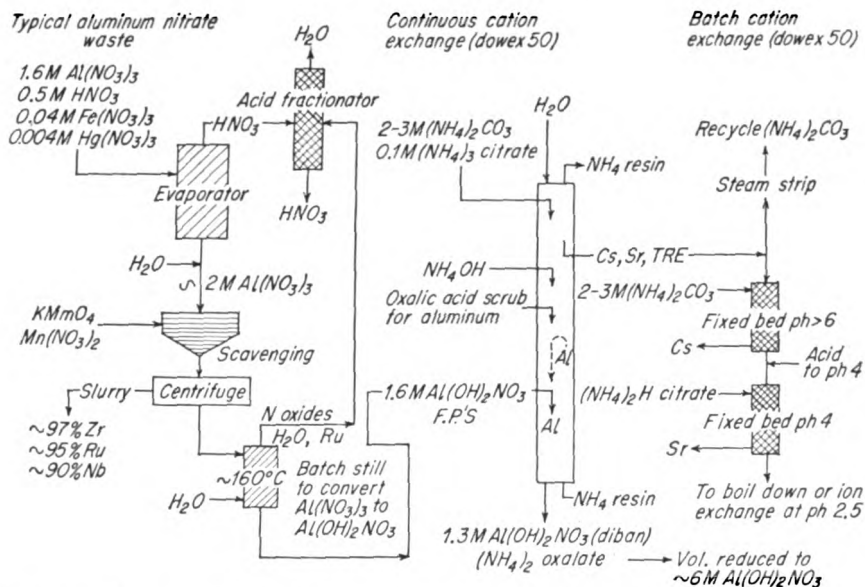


FIG. 21-4. Proposed scheme for removing fission products from radioactive wastes containing $\text{Al}(\text{NO}_3)_3$. (Higgins, Oak Ridge Natl. Lab. Rept. ORNL-1984, 1955.)

I. R., and R. G. Wymer, Diban—Ion Exchange Waste Disposal Scheme, I, *Oak Ridge Natl. Lab. Rept. ORNL-1984*, Nov. 28, 1955). The flow sheet for the scheme is shown in Fig. 21-4. The outstanding features of the process are:

1. Scavenging of the waste with $\text{Fe}(\text{OH})_3 \cdot \text{MnO}_2$ after evaporation of excess acid.
2. Conversion of $\text{Al}(\text{NO}_3)_3$ to dibasic aluminum nitrate $\text{Al}(\text{OH})_2\text{NO}_3$, in which aluminum has the characteristics of a monovalent ion.
3. Stripping the aluminum from the waste in a continuous ion-exchange contactor, thus minimizing the effect of the aluminum on the removal of fission products.
4. Removal of fission products from the resin after a minimum contact time to decrease radiation damages to the exchange material.
5. Concentration of fission products by batch-column treatment.

ADSORPTION ON CLAY

The adsorptive properties of clays are important not only as a means of concentrating radioactive wastes under controlled conditions but also because of the possibility that adsorption on the muds and silts at the bottom of natural rivers will create local hazards by the **uncontrolled concentration of radioactive wastes**.

The **exchange properties of clays** are due to the presence of fine-grained crystalline inorganic materials which make up the clay minerals, among which are found montmorillonite, hydrous mica, and kaolinite. These minerals are platy in physical structure, being composed of planes of Si ions surrounded by four O ions in the form of a tetrahedron, and planes of Al, Fe, or Mg ions surrounded by six O or OH ions

Table 21-13. Removal of Radioactive Materials by Conasauga Shale * (Jar-stirring Method) †

Jar No.	Shale, g	% removals								W-8 ‡	Con-tact time, hr
		Ba ¹⁴⁰	Ce ¹⁴⁴	Cs ¹³⁷	I ¹³¹	P ³²	Ru ¹⁰⁶	Sr ⁹⁰	Zr ⁹⁶		
1	0.5	83.4	79.0	97.7	1.8	33.7	80.4	40.0	87.6	60.7	1
2	1.0	91.8	98.0	98.2	3.6	46.7	93.0	39.8	97.2	68.3	2
3	2.0	88.0	71.6	97.9	3.4	42.4	83.0	44.4	84.7	65.7	1
4	3.0	94.7	98.3	98.6	9.9	69.3	97.8	40.7	97.6	73.7	2
5	5.0	89.3	78.4	97.5	8.0	66.0	87.8	47.2	85.7	69.6	1
6	10.0	96.4	97.2	98.4	9.3	82.2	98.3	45.1	98.3	72.4	2
7	12.0	93.5	95.5	98.8	12.8	68.1	90.9	59.9	89.8	77.5	1
8	15.0	97.5	99.7	99.2	20.3	78.2	99.5	54.4	98.6	79.5	2
9	18.0	94.5	98.0	99.1	18.0	66.0	94.6	63.8	91.8	79.2	1
10	20.0	98.2	98.9	99.2	25.3	78.7	99.5	64.1	99.2	80.4	2
pH.....		97.2	99.9	99.0	26.2	71.2	95.7	62.2	90.3	83.6	1
Range.....		99.3	99.9	99.2	39.0	81.1	99.5	73.6	98.6	82.5	2
		98.0	99.8	99.0	44.3	83.3	97.2	71.4	93.7	80.9	1
		99.5	99.9	99.2	38.6	95.7	99.6	71.9	99.4	81.6	2
		98.1	99.9	99.2	44.5	77.2	99.7	76.5	91.8	83.2	1
		99.7	99.9	99.5	50.1	87.6	99.5	76.9	98.7	84.0	2
		98.6	99.9	99.5	48.3	95.3	97.8	80.4	97.3	82.4	1
		99.8	100.0	99.5	55.4	98.8	99.7	78.0	99.5	82.5	2
		99.0	100.0	99.6	57.2	95.7	97.3	83.7	98.7	85.2	1
		99.8	100.0	99.7	58.7	98.7	99.8	76.8	99.8	85.1	2
		6.8	6.7	6.1	6.6	6.3	6.5	6.3	6.6	4.5	
		7.9	8.0	7.6	8.1	8.1	7.9	8.0	8.0	6.7	

* 500 ml tap water, level of contaminant $\sim 3.24 \times 10^{-2} \mu\text{c/ml}$ or 7,200 curie/min/ml at 10% geo.

† Brockett, T. W., *Proc. Eighth Ind. Waste Conf., Purdue Univ.*, 1953.

‡ Three-year-old mixed fission-product solution containing approximately 16% Ce, 20% trivalent rare earths, 2% Ru, 20% Sr, 21% Cs, and 21% unknown.

§ Waste from W-8 waste-storage tank.

Table 21-14. Removal of Radioactive Materials by Conasauga Shale (Column * Method) †
(Flow rate: ~ 25 ml/hr)

Material	Level of contamination, mc/ml	% removal per volume through column, ml																					
		50	100	200	300	400	500	600	700	800	900	1,000	1,100	1,200	1,300	1,400	1,500	1,600	1,700	1,800	1,900	2,000	
Ba ¹⁴⁰	3.86 × 10 ⁻²	100	...	100	...	99.99	...	100	...	100	...	100	...	100	...	100	...	100	...	100	...	100	100
Cs ¹³⁴	1.62 × 10 ⁻¹	99.99	...	100	...	100	...	100	...	100	...	100	...	100	...	100	99.99	...	...	...	...	100	100
Cs ¹³⁷	9.1 × 10 ⁻²	100	...	100	...	100	...	100	...	100	...	99.99	...	100	...	100	100	...	99.99	...	...	100	...
I ¹³¹	3.2 × 10 ⁻²	...	87.3	...	80.2	81.0	77.0	80.6	...	...	73.6	80.2	...	73.4	...	...	100	...	88.2	72.8	...	65.3	...
Fe	1.78 × 10 ⁻²	...	96.8	...	...	99.8	96.8	99.8	...	99.8	...	99.8	...	99.8	...	99.8	99.8	...	99.8	99.8	...	...	...
Ru ¹⁰⁶	3.6 × 10 ⁻²	...	99.9	...	...	99.9	99.9	99.9	...	99.9	...	...	...	...	...	...	99.9	...	79.5	99.8	...	...	...
Sr ⁹⁰	5.59 × 10 ⁻²	98.2	95.2	...	...	...	...	...	...	80.6	...	...	...	...	...	...	79.8	79.6	79.5	79.5	...	...	...
Zr ⁹⁵	2.28 × 10 ⁻²	...	100	...	...	100	100	100	...	...	100	...	100	...	100	100	100	...	100	100	...	...	...
MFP ‡	3.52 × 10 ⁻²	...	100	99.9	99.9	99.9	99.9	99.9	99.9	99.9	99.9	99.9	99.9	99.9	99.9	99.9	99.9	99.9	99.9	99.9	99.9	99.9	99.9
MFP ¶	3.61 × 10 ⁻¹	...	...	99.9	99.9	...	...	99.9	99.9	99.9	99.8	...	...	99.6	...	96.8	...	...	92.2	93.6	93.5	93.2	...
W-8 §	4.27 × 10 ⁻²	98.5	...	93.2	...	...	...	...	88.9	89.4	88.7	...	88.4	...	...	...	85.8	...	...	85.7	...	...	...

* Lucetroid column 1 1/4 in. diameter by 6 in. long containing 75 g local Conasauga shale sieved to 60 to 70 mesh.

† Brockett, T. W., *Proc. Eighth Int. Waste Conf., Purdue Univ.*, 1963.

‡ Mixed fission products—same as jar-test mixture, 50 g Conasauga shale—60 to 70 mesh. Flow rate 100 ml/hr

¶ Mixed fission products—same as jar-test.

§ Waste solution from W-9 waste-storage tank.

Table 21-15. Removal of Radioactive Materials by Various Clays (Jar-stirring Method with 1 G of Clay) *

Sample No.	Type of clay and Locality	% removals after 2 hr contact						
		Ce ¹⁴⁴ 3.9 × 10 ⁻² μc/ml	P ³² 3.5 × 10 ⁻² μc/ml	Ru ¹⁰⁶ 4.8 × 10 ⁻² μc/ml	Sr ⁹⁰ 1.1 × 10 ⁻² μc/ml	Sr ⁹⁰ 4.1 × 10 ⁻² μc/ml	Zr ⁹⁵ 0.78 × 10 ⁻² μc/ml	MFP † 4.1 × 10 ⁻² μc/ml
1	Kaolinite Mesa Alta, N. Mex.	92.50	51.58	69.39	30.40	46.25	98.08	70.87
2	Halloysite Ureka, Utah	91.82	72.90	77.95	26.15	61.63	98.32	89.77
3	Montmorillonite Little Rock, Ark.	69.21	88.27	64.40	63.03	74.50	97.18	94.70
4	Montmorillonite Cameron, Ariz.	89.00	31.89	52.89	57.44	71.35	91.05	92.44
5	Montmorite Manito, Wash.	85.16	44.56	60.77	29.26	71.49	92.25	94.75
6	Metabentonite Tazewell, Va.	76.05	68.28	53.61	16.10	53.46	83.59	86.20
7	Metabentonite High Bridge, Ky.	88.20	80.05	68.08	41.75	59.87	97.13	86.16
8	Conasauga Shale Oak Ridge, Tenn.	90.91	72.67	71.69	14.15	55.69	96.63	86.88
9	Vermiculite (nonexpanded) Sylva, N. C.	95.94	60.15	68.69	26.77	65.53	93.87	

* Brockett, T. W., *Proc. Eighth Ind. Waste Conf., Purdue Univ.*, 1953.

† Mixed Fission Products ~16% Ce¹⁴⁴, ~20% trivalent rare earths, ~2% Ru¹⁰⁶, ~20% Sr⁹⁰, 21% Cs, ~21% unknown.

which form an octahedron (Kelley, W. P., "Cation Exchange in Soils," Reinhold, New York, 1948). The **colloidal** (and adsorptive) **properties of clays** are determined by particle size, the **porosity of the clay structure** (and its effect on the specific surface area), and the net charge carried by the particle in contact with a solvent. This charge can arise when loosely bound atoms within the crystal become ionized, or when free valences are accidentally present at the edges of broken crystals. Differences in the adsorptive properties of the different clay types are largely the result of differences in the influence exerted by each of the above factors. In most cases, more than one of the factors contributes in defining the over-all characteristics of the particle. The low adsorptive power of **kaolinite** is due to its low porosity and lack of exchangeable ions. On the other hand, clays of the montmorillonite group combine great porosity with a high inherent charge, the result of which is a clay of high adsorptive capacity.

The **montmorillonite clays** are unique in that the spacing of the platelets forming the particle is a function of the moisture content of the mixture. At high moisture contents the distance between the layers may approach 20 to 22 Å. This feature makes it possible for ions to diffuse readily within the crystal and to exchange with more loosely bound ions.

Studies on the adsorption of various fission-product isotopes by different clay types have led to the accumulation of a large body of data. The results of these studies, which include both batch tests and continuous column studies, are given in Tables 21-13 to 21-15 (Brockett, T. W., Jr., and O. R. Placak, Removal of Radioisotopes from Waste Solutions by Soils—Soil Studies with Conasauga Shale, *Proc. Eighth Ind. Waste Conf., Purdue Univ.*, 1953. Straub, C. P., and H. L. Krieger, Removal of Radioisotopes from Waste Solutions—Soil Suspension Studies, *Proc. Eighth Ind. Waste Conf., Purdue Univ.*, 1953).

EVAPORATION

Evaporation is the most effective method in use for separating active wastes into a concentrated fraction and an essentially uncontaminated dilute effluent. However, because of the costs involved, evaporation is used only when essentially complete decontamination is required. A number of systems have been in use, which are described below.

A simple system (Browder, F. N., Liquid Waste Disposal at Oak Ridge National Laboratory, *Ind. Eng. Chem.*, vol. 43, pp. 7, 1502-1505, 1951. Nicholson, E. L., Design and Initial Operation of the Radiochemical Waste Evaporator, *Oak Ridge Natl. Lab. Rept. ORNL-393*, 1949) has been used at the Oak Ridge National Laboratory for the primary purpose of concentrating waste liquid held in underground storage tanks to achieve more storage volume. This system consists of a simple pot-still evaporator fed from a 2,000-gal evaporator storage tank. The still pot is heated by six removable steam coils. The flow of waste into the evaporator is controlled by the height of liquid in the still pot. Carry-over in the vapor leaving the still is separated in cyclone separators. Condensate flows first to a catch tank to be monitored and then is discharged to a large settling basin (Fig. 21-5).

Because of the high background in the vicinity of the evaporator, entrainment (and carry-over of radioactivity) is detected by a conductivity probe in a small

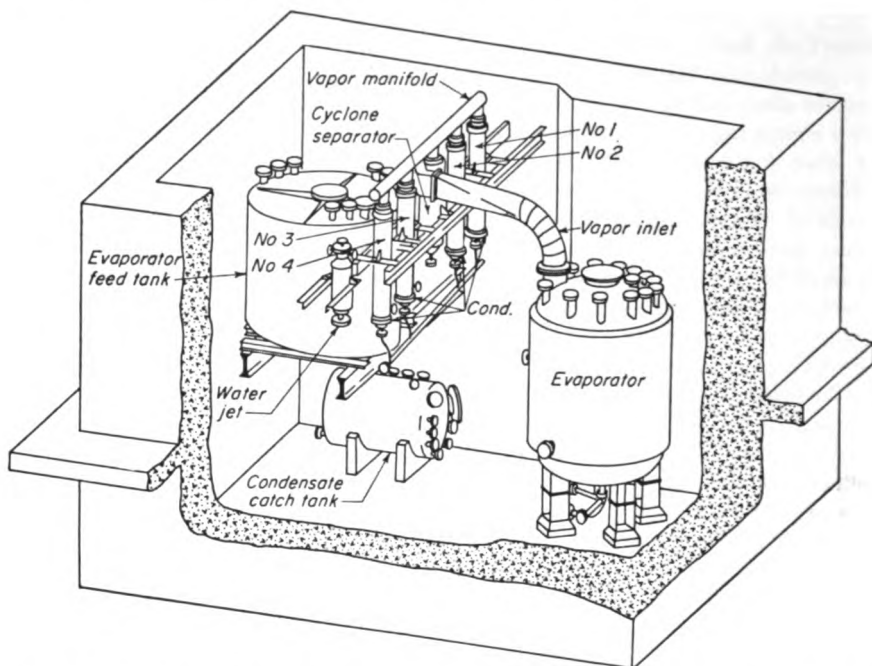


FIG. 21-5. Radiochemical waste evaporator. Piping not shown. (Nicholson, Oak Ridge Natl. Lab. Rept. ORNL-393, 1949.)

catch tank located immediately following the condenser. The cell actuates an alarm to notify the operator to reduce the water level in the still pot or to lower the steam pressure.

The end of a run is determined by the specific gravity of the liquid in the still pot, which is allowed to reach a value of about 1.3.

The Oak Ridge evaporator was designed for an operating capacity of 300 gal/hr but during initial operation was operated at about 600 gal/hr because of the need for additional tank-storage space. For this reason, a high decontamination factor was frequently sacrificed in order to achieve rapid reduction in volume. During this period the decontamination factor averaged 2.94×10^3 , and a volume reduction of 60 to 1 was attained. Use of this evaporator was discontinued after pit disposal of liquid wastes was initiated.

Combined wastes collected at the Knolls Atomic Power Laboratory (McCullough, G. E., Concentration of Radioactive Liquid Wastes by Evaporation, *Ind. Eng. Chem.*, vol. 43, pp. 7, 1505-1509, 1951. Glasson, T. J. E., et al., Costs of Waste Processing by Means of Evaporation at the Knolls Atomic Power Laboratory, KAPL-594, 1951) are collected and blended in 10,000-gal tanks before being fed to one of two evaporators in the system diagrammed in Fig. 21-6. The waste may contain 1.2×10^4 disintegrations/min/liter of alpha activity and 1.4×10^6 disintegrations/min/liter of beta-alpha radiation. The pH of the feed is adjusted with 5 per cent NaOH to 7.5 to 8.5 before entering the evaporator.

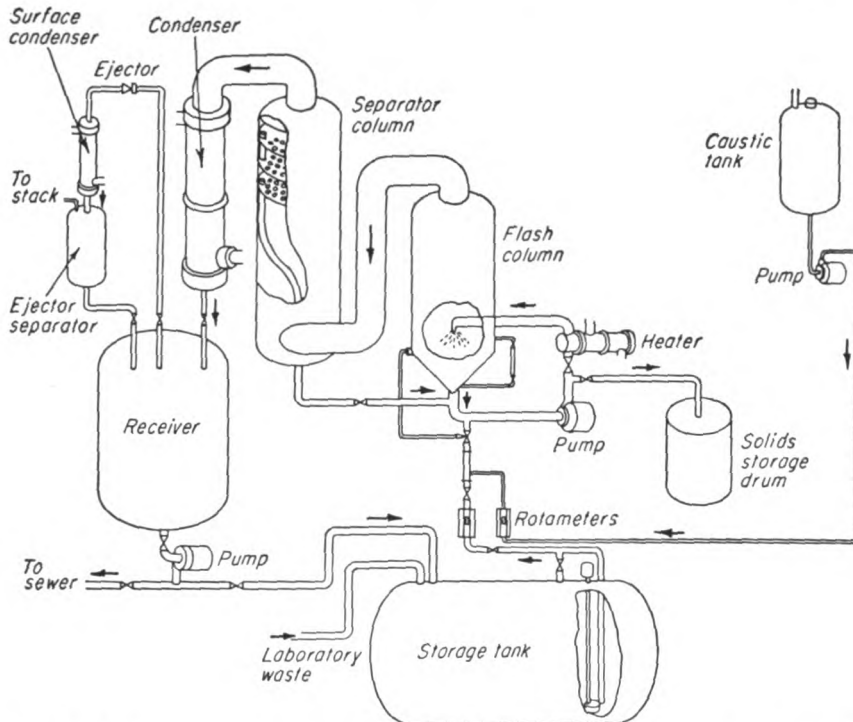


FIG. 21-6. Evaporator system—Knolls Atomic Power Laboratory. (Glasson, KAPL-594, 1951.)

The evaporator, designed for handling 500 gal/hr of waste, is a **forced-circulation system** consisting of a flash pan, pumps, heat exchanger, separating tower, and condenser. The flash pan is 6 ft in diameter by 12 ft high and contains about 300 gal of liquid under normal operating conditions. The liquid is recirculated through a heat exchanger at the rate of 650 gal/min and back into the flash pan. Roughly, 1 per cent of the liquid entering the flash pan, which operates under a vacuum of 26 in. of mercury, flashes into vapor.

The vapor passes through baffles which separate the larger drops from the stream and into the 6-ft-diameter by 20-ft-high separating tower. Entering tangentially, the vapor flows through a 12-ft deentrainment section and finally through four bubble trays located in the top section of the tower. The vapor is condensed and is caught in one of two 5,000-gal tanks to be monitored. Condensate containing less than 300 disintegrations/min/liter of alpha activity and less than 4,000 disintegrations/min/liter of beta-alpha activity is discharged to the storm sewer.

Condensate not meeting these requirements may be returned to the evaporator storage tank for retreatment. In normal operation, the condensate averages 0 to 2 (± 4) disintegrations/min/liter of alpha and 2 to 35 (± 20) disintegrations/min/liter of beta-alpha activity.

After the evaporator heel reaches about 20 per cent solids, the feed is stopped and the heel further concentrated to approximately 40 per cent solids. Approximate costs for evaporation by this system are shown in Table 21-16.

Table 21-16. Costs of Evaporation at Knolls Atomic Power Laboratory *
(Based on two-shift single-evaporator operation)

Direct costs—labor, materials, power, steam, laboratory analyses, maintenance.....	\$0.029/gal
Indirect laboratory expenses.....	0.037
Amortization—building (40 years), equipment, including installation (10 years).....	0.072
Total.....	\$0.138/gal

* Glasson, T. J. E., *et al.*, Costs of Waste Processing by Means of Evaporation at the Knolls Atomic Power Laboratory, KAPL-594, 1951.

A schematic arrangement of the **vapor-compression evaporation** system used at the Brookhaven National Laboratory is shown in Fig. 21-7 (Manowitz, B., *et al.*, Vapor Compression Evaporation Handles Radioactive Waste Disposal, *Chem. Eng.*,

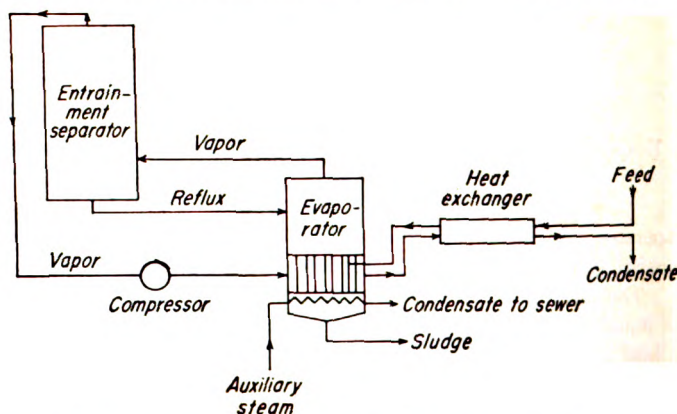


FIG. 21-7. Flow diagram for vapor-compression evaporator system. (Manowitz, B., *et al.*, Vapor Compression Evaporation Handles Radioactive Waste Disposal, *Chem. Eng.*, vol. 62, no. 3, pp. 194-196, 1955.)

vol. 62, pp. 3, 194-196, 1955). With this type of system, steam economy is maintained at a high level by compressing the vapor leaving the deentrainment device and feeding it into the steam chest of the evaporator as a source of heat. Auxiliary heat is supplied by steam fed into a coil located in the still pot beneath the evaporator calandria, which in the case of the Brookhaven evaporator, consists of 894 seven-eighths-inch outside diameter by 16-gauge stainless-steel vertical tubes. The liquor boils inside the tubes, and the resulting spray which is thrown up is deflected downward by a baffle located above the top tube sheet.

The deentrainment device consists of a 3-ft bed of 14- to 20-micron Fiberglas packed to a density of 5 lb/cu ft. Reflux from condensed vapor and liquid keeps the glass washed down. Steam leaving the separator is compressed and feeds into the shell side of the tube bundle. Condensate from the shell flows through a heat exchanger used to preheat the raw waste.

The length of a run is determined by the level of radiation external to the still or by the pressure differential under which the compressor may operate satisfactorily. The sludge is pumped to a drumming station where it is mixed with cement in the ratio of 25 gal of sludge to 384 lb cement. After the cement has hardened and any liquid has been siphoned off for reevaporation, the drums are sealed and buried at sea.

Operating characteristics and costs are given in Table 21-17.

Table 21-17. Operating Characteristics and Cost of Brookhaven Evaporator *

Design rate, 300 gal/hr	
Specific gravity of feed, 1.01 or less	
pH of feed, 7 (controlled)	
Steam economy, 15-16 lb distillate per 1,000 Btu input	
Sludge volume, 1.3% of feed	
Radioactivity in feed, up to 3×10^{-7} curie/ml	
MPC of effluent, 3×10^{-12} curie/ml	
Decontamination factor, 10^6 - 10^7	
Auxiliary steam, 60 lb/hr at 15 psig	
Cost:	
Evaporator, vapor, filter, and related equipment.....	\$ 60,000
Piping, valves, instrumentation.....	44,000
Design and field engineering.....	50,000
Building and installation.....	113,000
Operating costs (labor, supervision, utilities, drums)....	\$1,100/month

* Manowitz, B., *et al.*, Vapor Compression Evaporation Handles Radioactive Waste Disposal, *Chem. Eng.*, vol. 62, pp. 3, 194-196, 1955.

The evaporator used at Argonne National Laboratory (Rodger, W. A., and P. Fineman, Radioactive Waste Disposal, *Chem. Eng.*, vol. 58, no. 12, pp. 146-150, 1951. Rodger, W. A., P. Fineman, and H. Gladys Swope, Disposal of Radioactive Wastes at Argonne National Laboratory, *Proc. Eighth Ind. Waste Conf., Purdue Univ.*, pp. 474-491, 1953. Rodger, W. A., and P. Fineman, A Complete Waste Disposal System for a Radiochemical Laboratory, *Nucleonics*, vol. 9, no. 6, pp. 51-61, 1951) is a vertical-tube **natural-circulation evaporator** fabricated from type 316 stainless steel. It consists of a steam chest, separator chamber, overhead condenser, and constant-head feed tank. The steam chest is 18 in. in diameter and contains fifty-eight 5-ft-long 1½-in. BWG-16 tubes providing a total heating surface of 106 sq ft. The boil-up rate with a feed of about 0.08 per cent solids and a steam pressure of 15 psig is 150 gal/hr. The vapor and entrained liquid leave the evaporator through a centrifugal separating section and enter a 20-in.-diameter by 13-ft-high deentrainment separator. Liquid caught in the separator is returned to the steam chest;

the vapor passes through a Centrifix scrubber and is condensed in a triple-pass surface condenser, from which it flows to a catch tank for monitoring.

The waste is concentrated until the still-pot contents reach a solids concentration of about 45 per cent by weight. Radioactivity in the condensate has not exceeded 7×10^{-7} microcuries/ml of beta radiation. Steam economy equals 0.85 lb condensate/lb of steam. Costs for the Argonne evaporator are given in Table 21-18.

Table 21-18. Evaporation Costs, Argonne National Laboratory *

Equipment cost.....	\$14,400
Installation cost.....	3,500
	\$17,900
Annual operating cost:	
Steam at \$1.00/1,000 lb.....	\$1,200
Water at \$0.03/1,000 lb.....	460
Electricity at \$0.0083/kwhr.....	10
Labor and supervision.....	2,120
1 man-hr/hr labor at \$2.00/hr	
$\frac{1}{4}$ man-hr/hr supervision at \$2.40/hr	
Maintenance, 100% direct labor.....	1,640
Depreciation, 20%/year of total major equipment cost....	3,580
Overhead, 100% direct labor charge.....	1,640
Reagents.....	80
Building amortization, 10 year period.....	2,000
Cost to concentrate 116,000 gal of 0.08 weight per amount	
waste to 256 gal of 45 weight % solution.....	\$12,730
Cost per gal of feed processed.....	\$0.11/gal
Cost to concentrate 256 gal of 45% solution to 138 gal of	
80 weight %.....	0.01/gal
Total cost per gal.....	\$0.12
Basis:	
1. 1,200 gal/day through-put	
2. Steam economy, 0.85 lb condensate/lb steam supplied	
3. 18 days' operation with 1 day maintenance	
4. No reflux	
5. Steam used on days required for maintenance	

* Rodger, W. A., *et al.*, Disposal of Radioactive Wastes at Argonne National Laboratory, *Proc. Eighth Ind. Waste Conf., Purdue Univ.*, pp. 474-491, 1953.

ENTRAINMENT IN EVAPORATORS

Regardless of the system of evaporation employed, the over-all decontamination factor achieved is determined by the amount of carry-over in the vapor (Manowitz, B., R. H. Britton, and R. V. Harrigan, Entrainment in Evaporators, *Chem. Eng. Prog.*, vol. 51, no. 7, pp. 313-319, 1955). Carry-over is caused chiefly by splashing of the violently boiling liquid but may also be the result of entrainment of small

droplets or of foaming. Foaming in an evaporator can be controlled usually either by dilution of the contents of the still pot, by decreasing the boil-up rate, or by the addition of one of several antifoam agents. Control of entrainment and carry-over due to splashing is usually by means of devices to intercept the liquid directly, or devices which use gravitational or electrostatic forces to separate the liquid from the vapor. For large particles (~ 100 microns), bubble-cap plates, cyclone separators, or packed columns are effective. Smaller particles in the range of 10 microns or less must be separated by filters (Fiberglas or paper), or by electrostatic precipitators.

Table 21-19 gives the results obtained during the design of the Brookhaven evaporator on the efficiency of three types of deentrainment devices.

Table 21-19. Relative Efficiency of Three Deentrainment Devices *

Feed: fission-product solution
Boil-up rate: 10 to 65 lb/hr/sq ft

Device	Still-pot concentrate, Overhead concentrate	Relative efficiency, %
8.8 ft, $\frac{1}{2}$ -in. Raschig rings.....	9×10^5	2.2
13 plates, 2-in. bubble caps.....	6×10^6	15
4.5 ft, 20 microns Fiberglas.....	4×10^7	100

* Manowitz, B., R. H. Britton, and R. V. Harrigan, Entrainment in Evaporators, Chemical Engineering Progress, vol. 51, no. 7, pp. 313-319, 1955.

BIOLOGICAL PROCESSES

Extensive studies on the feasibility of treating radioactive wastes by **biochemical processes** have been made, particularly for use with laundry wastes which contain oxidizable organic complexing agents. In general, these processes have been found to have only limited possibilities. Since the radionuclides occurring in waste solutions are almost entirely inorganic chemicals, only those elements which are used nutritionally by the microorganisms will be concentrated in the cell structure. The concentration of other elements by living organisms can be only by surface phenomena which are subject to the laws of mass action and are influenced by attractive and repulsive forces due to charge. Since, in a waste, the radioactive constituents are usually present in only trace quantities, the effect of the bulk ion is usually sufficient to displace the trace element. In addition, elements basic to the metabolic functioning of the cell will be removed effectively only if the element is present in quantities sufficiently small to be the limiting factor for bacterial growth. If an excess of the element is present in stable form, the radioactive portion will be utilized only to the same extent as the nonradioactive portion.

Adsorption of radioactivity on the metallic parts of drains and sewers, or upon the slimes generally found in plumbing, may occasionally be sufficient to create a hazard to workmen. In a majority of cases it is possible to decontaminate such parts to a point where the hazard is below permissible limits. Nitric acid has been

found effective in removing strontium; alkaline solutions are effective for I-131 and P-32 (Talboys, A. P., Contamination of Plumbing by Low-level Radioisotope Wastes, *AEC Rept.* NYO-4010, May 1, 1952).

DISPOSAL OF LIQUID WASTES

Discharge and Dilution

Discharge and dilution of radioactive liquid wastes are widely practiced in the case of low-level wastes as brought out in the discussion on site location. For users licensed by the AEC, this method of disposal is acceptable provided that the rules and regulations specified by the Commission are followed. For discharge into a public sewerage system the regulations require that the following conditions be met.

1. The quantity released, if diluted by the average daily quantity of sewage released by the licensee, will not cause the average concentration to exceed the value for the maximum permissible concentration for that element listed in NBS Handbook 52. (See also Code of Federal Regulations, Title 10, Part 20.)

2. The quantity of material released in any one month shall not exceed that which would result in an average concentration in excess of the value given in NBS Handbook 52 if diluted with the average monthly flow of water released by the licensee.

3. The average concentration of any radioactive material released into an uncontrolled area shall not exceed, at the point at which the licensee loses effective control, the value given in NBS Handbook 52, or the Federal Register (See 1 above).

The limitations given above are subject to modification upon application by the user and approval by the Commission. Special rules have been approved for use by hospitals using radioactive phosphorus and iodine.

Discharge of liquid wastes to the environment from AEC laboratories has been practiced since the program began. The amounts discharged vary depending on the dilution available and on the proximity of the installation to centers of population and to other users of the receiving stream. A continuing program of monitoring is carried on to ensure that the permissible limits are not exceeded. Surveys on the aquatic life in the receiving stream have proved that, with the present levels of activity being discharged, no dangerous reconcentrations of radioactivity have occurred either by deposition of active material on silts or sand, or through concentration by microorganisms which may be consumed by fish and other higher forms.

Discharge from hospitals using the short-lived isotopes P-32 and I-131 are governed by the regulations in NBS Handbook 49, "Recommendations for Waste Disposal of Phosphorus-32 and Iodine-131 for Medical Users." The recommendations in this handbook are designed to protect both sanitation workers and sewage plant personnel. Unregulated discharge of P-32 and I-131 is permissible provided that the following conditions are met:

1. The transient, or short-period, concentration does not exceed 0.1 mc/liter of sewage.

2. A single-batch discharge of 10 mc/million gal of water flow is permissible.

3. Not more than 100 mc/million gal of sewage flow/day shall be discharged by the constant-drip-discharge method.

The batch-discharge method referred to above consists of collecting the active wastes in a 1-gal container, filling it to the top with water, and pouring into the drain. The "constant-drip-discharge method" consists of uniformly discharging the active waste (containing less than 100 mc/10⁶ gal of sewage/day) through a capillary into the drain at a constant rate over a 6-hr period.

The levels of contamination permitted in the sewerage systems have been computed on the basis of studies made on the concentration of radioisotopes by bacterial slimes and sewage microorganisms. In the case of I-131, concentration by microorganisms is quite limited. With P-32, there is considerable isotopic dilution by the stable phosphorus normally present in domestic sewage so that the concentration of P-32 in sewage sludges to dangerous levels cannot occur. In cases in which the level of contamination will exceed that permitted it is necessary to collect and store the waste for decay or for discharge by one of the methods described, but over a longer period of time. When P-32 and I-131 are used simultaneously in the same institution, disposal rules should be based on the sum of the activities of the two isotopes.

Storage of Radioactive Liquid Wastes

Storage of **low-level liquid wastes** is practiced only as a means of holding these wastes for monitoring or to permit the decay of short-lived isotopes such as P-32 or I-131. No shielding is required for this purpose. In many cases, tanks used for temporary storage of low-activity wastes are made of stainless steel, as the waste may contain high concentrations of acid.

High-level liquid-waste storage has the primary advantage of maintaining the waste under controlled conditions. Because of the costs involved, tank storage is considered to be a solution to the waste problem only until such time as other methods can be perfected. Although stainless-steel tanks are sometimes used in special applications the majority of tanks in use, primarily at Oak Ridge, Tenn., and Hanford, Wash., have been constructed of mild steel and reinforced concrete and have varied in size from 30,000 to 1,000,000 gal capacity. The tanks are buried in soil to provide adequate shielding.

Before being pumped into a storage tank the waste is made alkaline, which helps minimize corrosion and causes the metallic elements to precipitate as hydroxides which ultimately settle to the bottom of the tank.

A large amount of heat is generated within the waste through the disintegration of the fission products. If the rate of heat loss to the surrounding soil is exceeded by the rate of heat generation, the temperature within the tank will rise and the waste may eventually reach the boiling point. Under these conditions, if provision has been made to vent the escaping steam, the waste will be self-concentrating. Figures 21-8 and 21-9 (Anderson, C. R., *et al.*, Design and Operation of High Level Waste Storage Facilities, *Proc. Intern. Conf. Peaceful Uses Atomic Energy*, vol. 9, p. 640, 1956. Coppinger, E. A., *et al.*, Heat Problems in the Disposal of High Level Radioactive Wastes, *AEC Rept. TID-7517*, vol. 16, p. 475, 1956) show schematically a waste system designed to take advantage of the self-concentration effect.

Vapors from the tank are deentrained by a cyclone separator or similar device and then condensed in a water-cooled condenser. The entire system is vented to the atmosphere through a filter and stack. The condensate may be handled as a low- or intermediate-level waste. At the Hanford installation the condensate is permitted to percolate into the soil through a crib (see Disposal of Liquid Wastes to Pits and Wells).

In larger-sized tanks, or in tanks containing a layer of sludge, localized boiling known as bumping occurs because of a sudden release of heat which has been stored

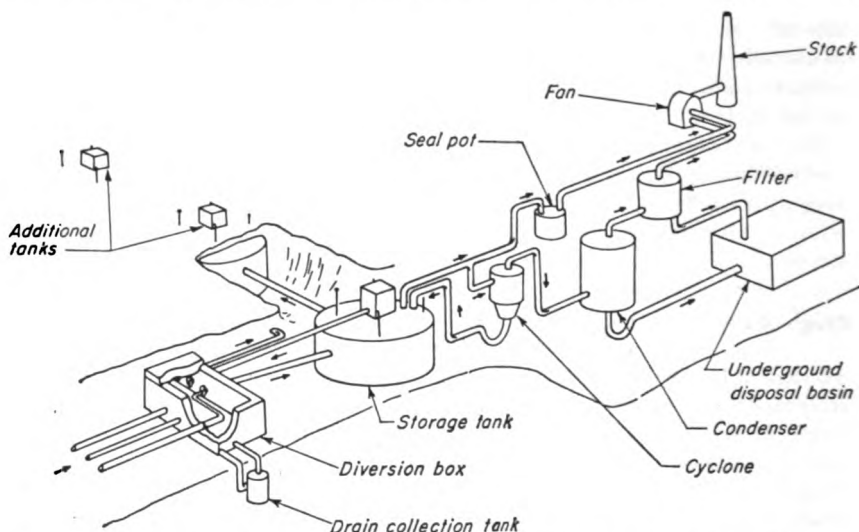


FIG. 21-8. Typical high-level waste-storage facility. (Coppinger, AEC Rept. TID-7517, 1956.)

in the waste solution as sensible heat. This action may continue for a considerable period of time and will contribute to the amount of deentrainment required. Tests have indicated that bumping can be minimized by agitating the liquid to produce a continuous updraft of liquid (Coppinger, E. A., and R. E. Tomlinson, Heat Problems in the Disposal of High Level Radioactive Wastes, AEC Rept. TID-7517, pt. 1b, October, 1956). The period of time over which a liquid waste will continue to boil is a function of the concentration of waste, the size of the tank, and the rate of heat loss to the surrounding soil.

A waste system can be designed to prevent the waste from boiling if this appears desirable. The tanks can be constructed with cooling coils to remove sufficient heat to keep the waste below the boiling temperature. Use of this system increases the cost of storage by the additional cost of the cooling system and by eliminating the added storage volume obtained through self-concentration. In addition, use of such a system requires that the cooling water be monitored to detect leaks in the system, and that venting and deentrainment facilities be provided in the event of a cooling-system breakdown. Boiling may also be prevented by diluting the waste sufficiently to decrease the amount of heat generated per unit volume, or by pro-

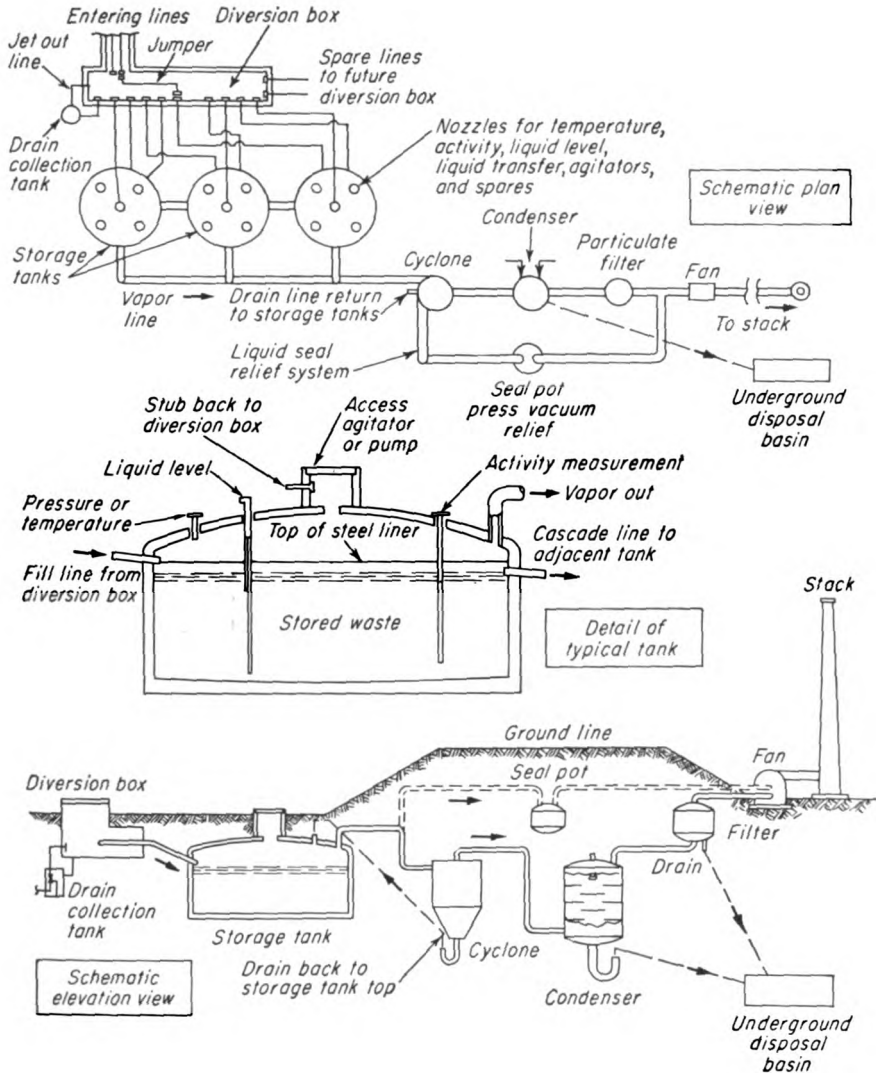


FIG. 21-9. Design and operation of high-level waste-storage facilities. (Typical high-level waste-storage facility.) (Anderson, C. R., et al., *Design and Operation of High Level Waste Storage Facilities*, Proc. Intern. Conf. Peaceful Uses Atomic Energy, Geneva, vol. 9, p. 640, 1956.)

viding a large number of small tanks to permit a large surface-to-volume ratio to expedite dissipation of heat to the soil. Any of these alternate methods is feasible but probably can be justified economically only if it can be shown that tanks using the self-concentrating feature are unsafe.

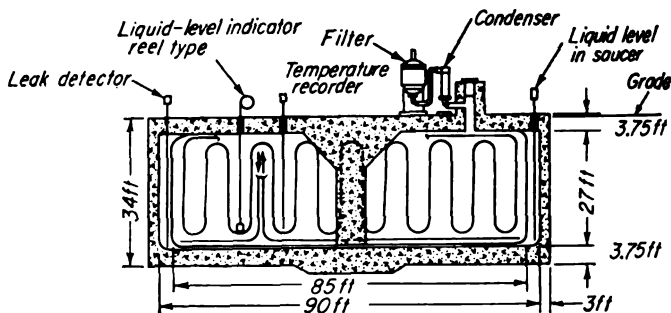


FIG. 21-10. Internally cooled waste-storage tank (elevation). 85 ft diameter by 27 ft high, 1,070,000 gal. (Wilson, E. E., *Design Considerations of Storage Tanks for Radioactive Wastes*, Chem. Eng. Progr. Symposium Ser., vol. 52, no. 19, Nuclear Eng., pt. IV, p. 157, 1956.)

In any system of tank storage a potential hazard exists from the accidental release of fission products to the environment, such as would occur from a rupture of the tank. Studies on the corrosion rate of the steel liners in presently used equipment

indicate that the potential life of these tanks is around 100 years. With proper design and construction, leaks within the system can be detected from a system of dry wells, or secondary containers surrounding the primary container and with sufficient capacity to hold the leakage until the tank contents can be transferred.

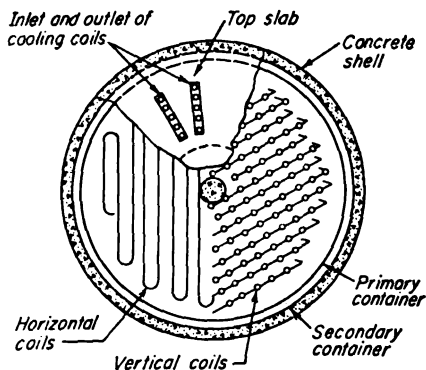


FIG. 21-11. Internally cooled waste-storage tank (plan view). 85 ft diameter by 27 ft high, 1,070,000 gal. (Wilson, E. E., *Design Considerations of Storage Tanks for Radioactive Wastes*, Chem. Eng. Progr. Symposium Ser., vol. 52, no. 19, Nuclear Eng., pt. IV, p. 157, 1956.)

from $\frac{5}{8}$ in. at the top to $\frac{3}{8}$ in. at the bottom. All welds were radiographed. The secondary container is also of $\frac{1}{2}$ -in. steel and is 90 ft in diameter by 5 ft high. The secondary container is a saucer-type container designed to hold a portion of

The design of a large-sized tank utilizing a secondary container and with interval cooling is shown in Figs. 21-10 and 21-11. (Wilson, E. E., *Design Considerations of Storage Tanks for Radioactive Wastes*, Chem. Eng. Progr. Symposium Ser., vol. 52, no. 19, Nuclear Eng., pt. IV, pp. 153-157, 1956.) The primary container of this tank (1,070,000 gal capacity) is 85 ft in diameter by 27 ft high. The steel top and bottom are $\frac{1}{2}$ in. thick. The cylindrical portion varies

the primary-tank contents in the event of a leak. The two containers are housed in a concrete shell 2 ft 6 in. thick at the bottom with 1 ft 10 in. thick walls and top. The housing is waterproofed by a five-ply asphalt-saturated membrane and by 3 in. of mastic poured between the housing wall and an outer course of brick.

Cooling is accomplished by two sets of coils, a horizontal set near the bottom to provide cooling while the tank is only partially filled, and a vertical set of hairpin coils to provide cooling when the tank is full.

Capital costs for storage facilities have been estimated to be between \$0.37 and \$2.00 per gallon, dependent on the type of facility constructed.

Ocean Disposal as Liquid or Solid

Recommendations for the ocean disposal of materials which can be encased in massive containers are given in NBS Handbook and have been summarized below:

1. Disposal of all packaged wastes shall be in regions where the water depth is greater than 1,000 fathoms. This rule applies to all wastes except small quantities of liquid wastes.
2. Containers for the waste shall be constructed so that:
 - a. The package cannot be broken easily, and will sink to the ocean floor without loss of contents.
 - b. The package and contents are free of voids.
 - c. The package has an average density of at least 1 to 2 g/cm³.
 - d. The package presents no external radiation hazard.
 - e. The package is easily handled.

It is further recommended that the package be marked in such a way that it can be identified not only as to its most hazardous isotope but also by date and origin.

Table 21-20 summarizes the practices used at various locations in preparing low-level wastes for disposal at sea. No high-level wastes have been disposed of in this manner (Morgan, J. M., Technical and Economic Aspects of Disposal of Radioactive Waste at Sea, Wash-275, 1954).

The discharge of liquid radioactive wastes through pipelines directly into the ocean is not recommended. This is because of the possible hazard resulting from incomplete mixing at the outfall and subsequent migration to the shore. Concentration of the radioactivity by sediments and microorganisms is also a possible consequence of discharging liquid wastes to shallow waters.

Bulk discharge of liquid wastes from barges is permissible under certain conditions. The liquid should be in a water solution having a specific gravity greater than 1.1 to facilitate mixing. The liquid should be discharged at a rate such that the initial mixing and dilution afforded by the turbulent wake will reduce the concentration of the waste to harmless levels. Dumping should be restricted to areas where the water depth is greater than 1,000 fathoms. Oily or greasy liquids which tend to float should not be discharged as liquids but may be packaged and discharged as a solid.

At the present time, only materials of rather low specific activity which can present no future hazard have been disposed of in ocean waters. That disposal of high-level wastes by discharge and dilution in the ocean would present tremendous

Table 21-20. General Practices for Preparing Metal Containers for Sea Disposal *

Installation	Solid waste †	Liquid waste ‡	Evaporator concentrate	Burial, routing and transshipping
Massachusetts Institute of Technology, Cambridge, Mass.	4-, 5-, 10-, 14-, and 40-gal cans and drums have been used. Wastes are placed in the containers and concrete poured on top to give sufficient weight to sink	Bottle contained liquids placed in c/s drums and concrete poured over and around	No evaporator waste	By MIT truck to Boston Navy Yard. U.S. Navy dumps in ammunition dumping area in Massachusetts Bay
Brookhaven National Laboratory, Upton, Lon	Used 55-gal c/s drums. 6 in. of concrete on bottom. Low-level wastes placed on this and a 4-in. cap provided. High-level wastes set into prepared concrete shell in drums	Used 55-gal drums. 6 in. of concrete on bottom. Liquid container placed on this and concrete poured over and around	New 55-gal c/s drums. 30 gal of concentrate poured into drum and 4 sacks of cement added. Capped drum rolled for mixing and allowed to set	By BNL truck to U.S. Navy Ammunition Depot, Earle, N.J. By U.S. Navy LST to dumping ground 120 mi into Atlantic. Burial 1,000 fathoms in ammunition-dumping area
Westinghouse Atomic Power Division, Pittsburgh, Pa.	Used 30- or 55-gal c/s drums. 4- to 8-in. concrete base. Mix waste and concrete to top of drum. Top of drum sealed with welded steel 24 in. in diam. Also ship-incinerated ashes using same method	No liquid waste shipped to sea	Used 30-gal c/s drums. Drum filled within 3 in. of top. Water-absorbent powder added if liquid seeps to top before sealing. 18-in. steel top welded to drum	By commercial truck to U.S. Navy Ammunition Depot, Earle, N.J. By U.S. Navy LST to dumping ground 120 miles into Atlantic. Burial 1,000 fathoms in ammunition-dumping area
California Research and Development, Livermore, Calif.	Used 30- or 55-gal c/s drums 4 to 8 in. concrete base. Mix waste and concrete to top of drum. Cables embedded in concrete with loop sticking out of open top used for ease in lifting	Used 55-gal c/s drums. 5-gal carboy set on concrete base and concrete poured around container. 1/4-in. wire cable also has been used for ease in lifting	No evaporator waste	By C.R.&D. truck to Hunters Point (USNRL) for shipment to Pacific Ocean in U.S. Navy Scow, about 60 miles west of San Francisco in 500 fathoms
University of California Radiation Laboratory, Berkeley, Calif.	Used 55-gal c/s drums. Waste is mixed with concrete throughout drum. Cable embedded in concrete with 1-ft loop sticking out of top used for lifting	Used 55-gal c/s drums. 5-gal carboy set on concrete base and concrete poured around container. Wire cable embedded in concrete used for ease in lifting	No evaporator waste	By University of California truck to Hunters Point (USNRL) for shipment to Pacific Ocean in U.S. Navy Scow, about 60 miles west of San Francisco in 500 fathoms
U.S. Naval Radiological Defense Laboratory, San Francisco, Calif.	Used 55-gal c/s drums. Waste is mixed with concrete by using alternating layers of waste and concrete. Outside sometimes shot-creted to reach minimum allowable weight. Open top	Used 55-gal c/s drums. 1 to 2 cu ft concrete bottom poured. Small and large lab containers placed on base. Drum capped with 6 to 8 in. concrete. Open top	No evaporator waste	Loaded on site to U.S. Navy Scow for disposal similar to above

* Morgan, J. M., Wash-275, 1964.

† Solid wastes include contaminated metal scraps, wood, rubber gloves, klenex or wipes, animal carcasses, laboratory apparatus and/or furniture, pipe and ductwork, floor sweepings, etc. For the most part the radioactive contamination is very low.

‡ Liquid wastes include those from radiological or radiochemical laboratories, and from special experimental and/or pilot-plant operations, generally referred to as bench wastes. They are placed in the drums in glass or polyethylene bottles varying in size from about 5 oz to 5 gal.

problems can be seen from Table 21-21 giving the volumes of liquid required to dilute the 50-year accumulation of fission products to the presently acceptable levels (Rodger, W. A., Trends in Separations Process Development and Their Effects upon Waste Handling, *AEC Rept. TID-7517*, pt. 1a, 1956).

Table 21-21. 50-year Accumulation of Long-lived Isotopes and Required Dispersal Volumes *

Basis: 2.2×10^6 mw installed reactor capacity = 3 tons fission products/day

Isotope	Accumulated quantity in 50 years, curies	MPC ¶ in		Volume required to dilute to tolerance	
		Water, $\mu\text{c}/\text{m}$	Air, $\mu\text{c}/\text{m}$	Water, cu miles	Air, cu miles
Zr ⁹⁵	1.3×10^{11}	4×10^{-3}	4×10^{-7}	7.8×10^3	7.8×10^7
Ce ¹⁴⁴	1.1×10^{11}	4×10^{-2}	7×10^{-9}	6.6×10^3	3.8×10^9
Ru ¹⁰⁶	1.0×10^{11}	0.1	3×10^{-8}	2.4×10^2	8×10^8
Pm ¹⁴⁷	5.1×10^{10}	1	2×10^{-7}	12	6×10^7
Sr ⁹⁰	8.6×10^{10}	8×10^{-7}	2×10^{-10}	2.6×10^7	1×10^{11}
Cs ¹³⁷	8.1×10^{10}	1.5×10^{-3}	2×10^{-7}	1.3×10^4	9.7×10^7
Tc ⁹⁹ †	2.0×10^7	3×10^{-2}	3×10^{-6}	0.2	1.6×10^3
Pu ²³⁹ ‡	2.8×10^6	1.5×10^{-6}	2×10^{-12}	4.5×10^2	3.4×10^8

* Rodger, W. A., Trends in Separations Process Development and Their Effects upon Waste Handling, *AEC Rept. TID-7517*, pt. 1a, 1956.

† Decay neglected.

‡ Based on a loss of 0.1% in processing.

¶ From NBS Handbook 52, 1953.

Disposal of Liquid Wastes to Pits and Wells

Under certain conditions, **ground disposal** of intermediate-level wastes is feasible and has been practiced at AEC installations in Washington and at Savannah River. Disposal of high-level wastes to open pits has been practiced at the Oak Ridge installation since 1951.

Because of favorable climatic and geologic conditions at the Hanford, Wash., site, considerable experience has been gained in the disposal of wastes to the ground through **cribs** constructed for this purpose, and in **open trenches** and ponds which can be backfilled after use. These systems for disposal are diagrammed in Fig. 21-12 (Brown, R. E., *et al.*, Disposal of Liquid Wastes to the Ground, *Proc. Intern. Conf. Peaceful Uses Atomic Energy*, Geneva, vol. 9, pp. 669-675, 1956).

Factors which influence the rate of movement and adsorption of liquid wastes which must be considered in a ground-disposal program include the following.

1. Chemical, physical, and radiochemical properties of the waste
2. The allowable concentration in public water supplies of each isotope in the waste
3. The effectiveness and permanence of retention of the active material on the soil

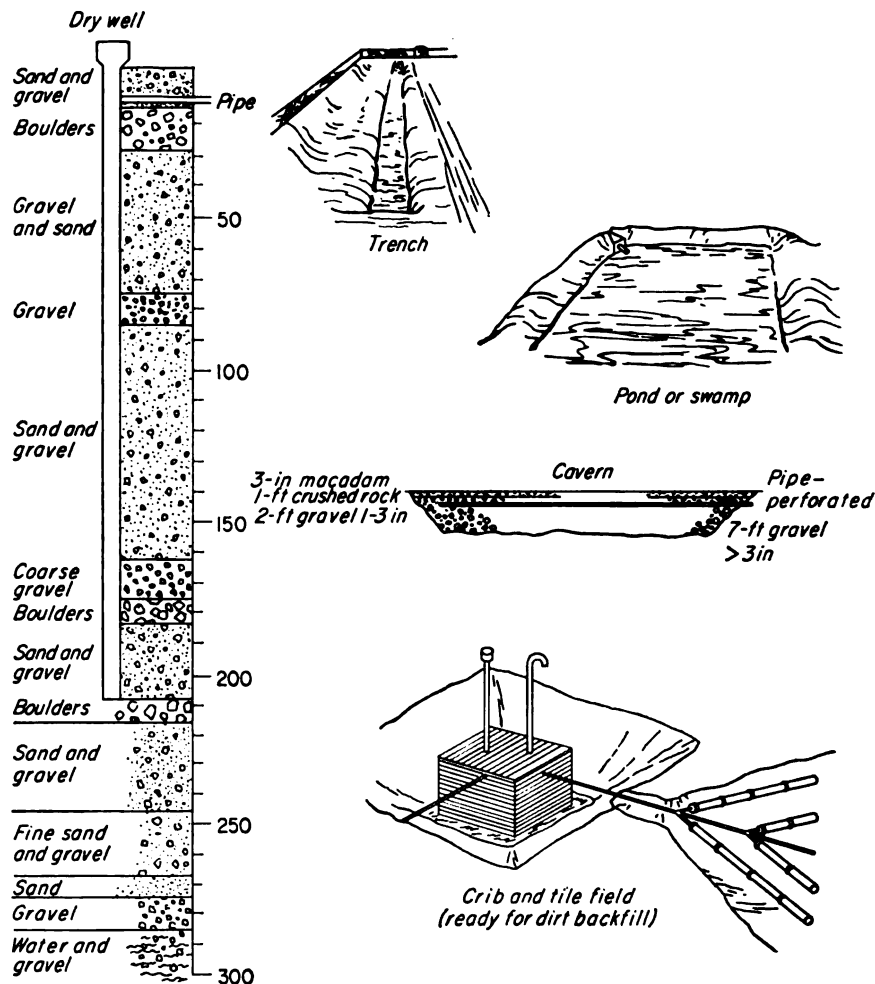


FIG. 21-12. Disposal of liquid wastes to the ground. (Brown, R. E., et al., *Disposal of Liquid Wastes to the Ground*, Proc. Intern. Conf. Peaceful Uses Atomic Energy, Geneva, vol. 9, pp. 669-675, 1956.)

4. The possibility that the adsorbed material may be leached from the soil by rain, or displaced by ions added in subsequent wastes

5. The location and direction of movement of the ground water and the amount of dilution afforded by the ground water.

The adsorption of various ions on Hanford-type soil is shown in Figs. 21-13 and 21-14. Studies have shown that the adsorbed ions can be leached completely from the soil by acids and by high concentrations of inert salts. The effect of NaNO_3 on the adsorption of cesium is shown in Fig. 21-15. A typical distribution of isotopes in the soil column through which the waste has passed is illustrated in Fig. 12-16.

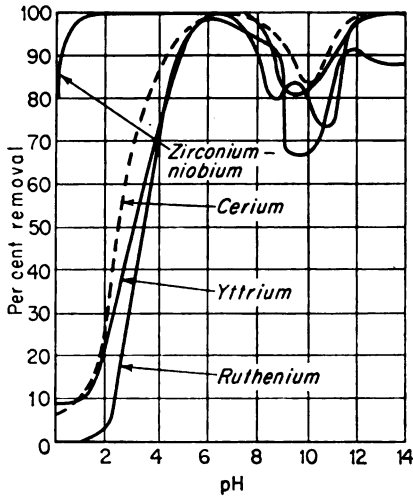


FIG. 21-13. Equilibrium adsorption of Zr, Ce, Y, and Ru on soil. (Brown, R. E., et al., *Disposal of Liquid Wastes to the Ground*, Proc. Intern. Conf. Peaceful Uses Atomic Energy, Geneva, vol. 9, pp. 669-675, 1956.)

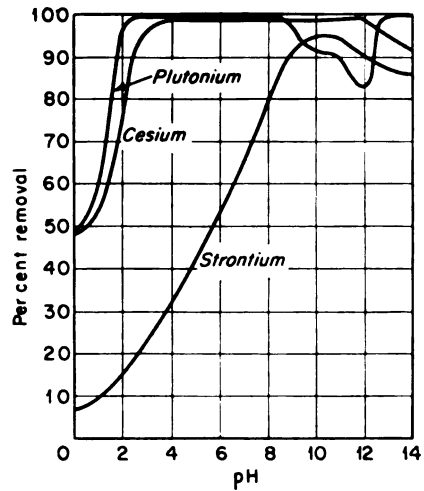


FIG. 21-14. Equilibrium adsorption of Pu, Cs, and Sr on soil. (Brown, R. E., et al., *Disposal of Liquid Wastes to the Ground*, Proc. Intern. Conf. Peaceful Uses Atomic Energy, Geneva, vol. 9, pp. 669-675, 1956.)

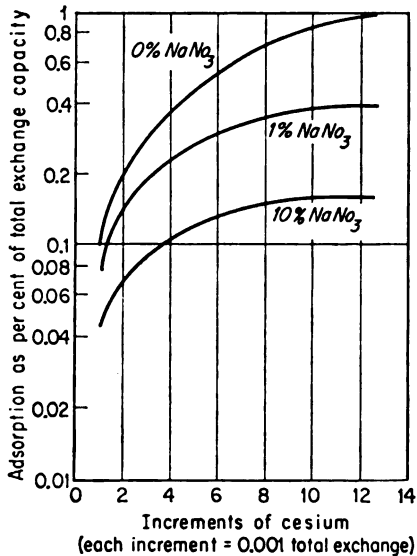


FIG. 21-15. Influence of NaNO_3 on adsorption of cesium by a Hanford soil. (Brown, R. E., et al., *Disposal of Liquid Wastes to the Ground*, Proc. Intern. Conf. Peaceful Uses Atomic Energy, Geneva, vol. 9, pp. 669-675, 1956.)

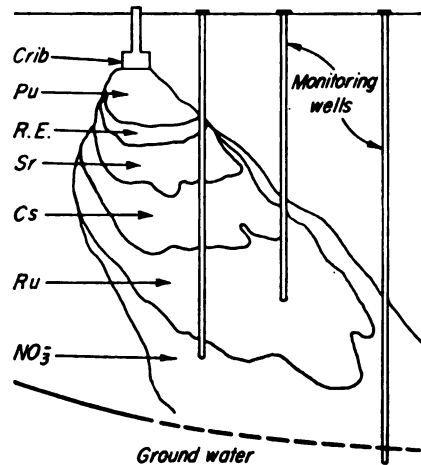


FIG. 21-16. Typical distribution of radioisotopes in Hanford soil. (Brown, R. E., et al., *Disposal of Liquid Wastes to the Ground*, Proc. Intern. Conf. Peaceful Uses Atomic Energy, Geneva, vol. 9, pp. 669-675, 1956.)

Ruthenium is found to be among the most mobile, probably because of its tendency to form nonionic and anionic species. Plutonium and strontium, both of biological significance, are held rather high in the column. Monovalent cesium is lower in the column because it is less firmly bound and is easily displaced by ions of higher valence.

At the Savannah River Plant, approximately 130,000 gal/day of low-specific-activity waste are discharged to the ground (Clark, J. R., and H. A. Pohl, *Earth Disposal of Radioactive Wastes at SRP, AEC Rept. TID-7517*, pt. 1a, 1956). These wastes are discharged into pits dug into the ground and are then permitted to seep into the soil. As the liquid moves through the soil the radioactive material becomes adsorbed. Since the levels of activity discharged are extremely low, the migration of the radioactivity through the soil column is very slow. In effect, the ground is serving as a storage area for the radioisotopes. The cost of the system capable of handling 130,000 gal/day was \$52,000.

Open-pit disposal of radioactive wastes was put into practice at the Oak Ridge National Laboratory after the tank system became incapable of handling the volume of waste produced. Four pits with a total volume of some 3,000,000 gal have been constructed in bare soil in an isolated section of the controlled area of the laboratory (Witkowski, E. J., *Operating Experience in the Disposal of Radioactive Wastes in Open Pits, AEC Rept. TID-7517*, pt. 1a, 1956). More than 2,700,000 gal of waste, containing more than 35,000 curies of radioactivity, has been pumped into the pits. Construction costs for the three 1,000,000-gal pits and the pipeline and pumping station were \$78,000.

The pits are open to the atmosphere but are covered with wire screen to prevent contamination of wildlife. Although air samples indicate no air-borne hazard from the radioactivity, the levels of radiation are too high to permit prolonged work in the immediate area. Considerable seepage from the pits has been observed due to faults in the geologic formations in the Oak Ridge area. The primary radioisotope found in the seepage is Ruthenium-106. Since the MPC of this isotope is high, and since the ground water flows into a lake and river affording high dilution, no radioactive hazard to the public has resulted from the operation of the pits. Seepage of nonradioactive nitrate ion from the pits has resulted in high concentrations of this ion in the impounded areas of the lake affected by the pits.

The feasibility of storing high-level liquid wastes in natural or specially prepared **underground openings** or in **salt domes** is under investigation. Use of salt domes, either naturally occurring or developed especially to contain the wastes, appears feasible because of the widespread occurrence of these formations which would greatly decrease the expense of transporting the waste (Thurston, W. R., *Summary of Princeton Conference on Disposal of High Level Radioactive Waste Products in Geologic Structures, AEC Rept. TID-7517*, pt. 1a, 1956). Costs of developing these cavities have been reported to be (for storage of hydrocarbons) from \$3 to \$6 per barrel. Before this method can be used, studies must be made of the behavior of salt in contact with the waste, and of the heat-transfer properties of the structure.

Excavation in **shale formations** is a second possible relatively inexpensive method of underground disposal. However, leakage from this type of formation is much more of a problem. In addition, the stability of shale under the influence of different types of wastes has not been evaluated.

Injection of wastes into clay-bearing beds of low permeability is a possible method of disposal but is subject to many restrictions. In particular, fluctuations in the water table must be determined, as only areas with low water table could be used for disposal operations. Injection of wastes into clay formations for adsorption and fixation of the active material will, in all probability, require that the waste be pretreated to remove inactive bulk chemicals and other materials likely to precipitate and plug the formation. Such pretreatment would, of course, add to the cost of this method of disposal.

Disposal into **abandoned dry mines** does not appear to be a method likely to be used, since the probability of locating such a mine in an area where it would be of use is small.

Deep-disposal methods are particularly suited for areas in which the geologic formations occur as rock layers of high porosity separated by other layers of impervious material. In selecting sites for disposal by deep injection the following points should be considered (Thurston, W. R., Summary of Princeton Conference on Disposal of High Level Radioactive Waste Products in Geologic Structures, *AEC Rept. TID-7517*, pt. 1a, 1956):

1. The liquids containing the radioactive wastes should have a greater specific gravity when introduced into the reservoir than the liquids already present in the reservoir. This would insure that any leakage or migration would be downward, still deeper into the earth, and takes care of undetected fractures and seismic rupturing of the reservoir stratum.

2. The liquids should be injected underground, preferably where they will remain under essentially static conditions.

It has been proposed that, if the rate, direction, and distance of movement of the liquids in a particular bed are known accurately enough to assure ample decay-time for the radioisotopes, wastes might be injected into this bed; to many geologists, the time and expense involved in making the study to be able to give complete assurance of safety appear prohibitive.

3. The introduction of the fluids into the bottom of structural basins is one means of obtaining the static conditions required in point 2 above. This does not preclude the possibility of locating, by intensive study, other types of structures that would provide static conditions far removed from the biologic environment.

The bottom of a structural basin is that part closest to the crystalline basement and consists of the oldest formations in the stratigraphic sequence of that basin. In some areas the depth may be only 5,000 feet but in others it may be 20,000 feet or more. At the bottom of a structural basin the wastes in almost every case would be below zones of potable water or valuable hydrocarbons; at 10,000 feet the economic limit of conventional mining has been exceeded.

4. The distribution of the waste liquids within the reservoir formation should be monitored by appropriate observation wells. These wells would also serve as sources of brine needed to dilute the waste solutions to dissipate heat. The withdrawal of brine from the reservoir formations will facilitate the injection of waste solutions by lowering pressures, and at the same time eliminate the possibility of building up pressures capable of fracturing the structure.

5. Prior to the introduction of nuclear wastes into the reservoir, the problems of heat dissipation, clogging of reservoir pore spaces, and chemical reactions with the rock and fluids would have to be evaluated. The bores made to obtain samples can be part of the final drilling pattern so that the cost of sampling would not be excessive for a successful installation.

Fixation of High-level Wastes

Fixation on clays of high-level radioactive wastes is a promising method of ultimate disposal. While adsorption on soils is an inherent feature of disposal into the ground through seepage pits, it is not feasible to depend on this method to fix permanently the radioactivity of high-level wastes. For this purpose, special clays, particularly of the montmorillonite type, have been found useful. Dilute wastes

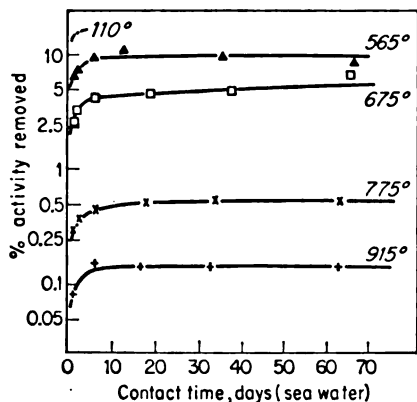


FIG. 21-17. Fixation of fission products on montmorillonite clay. (Hatch, L. P., *et al.*, *Processes for High Level Radioactive Waste Treatment, Proc. Intern. Conf. Peaceful Uses Atomic Energy, Geneva*, vol. 9, pp. 648-658, 1956.)

of low acidity and inert salt content can be adsorbed directly on extruded clay rods to achieve a concentration of several curies per gram of clay (Hatch, L. P., *et al.*, *Processes for High Level Radioactive Waste Treatment, Proc. Intern. Conf. Peaceful Uses Atomic Energy, Geneva*, vol. 9, pp. 648-658, 1956). Completeness of decontamination is a function of the composition of the waste and the length of the clay column. Ruthenium is poorly adsorbed, probably because of its tendency to form negatively charged complexes. After the clay is exhausted, it is fired at a temperature in the range of 900°C. This final step causes a change in the structural characteristics of the clay, which fixes the fission products in essentially unleachable form (Fig. 21-17).

Because of the high salt content of aluminum- and zirconium-bearing wastes (which interferes with the adsorption of the fission products) it is necessary to pretreat these wastes to minimize or eliminate the mass-action effect of the bulk ion. One proposal to accomplish this objective involves calcination of the aluminum nitrate waste in two stages (Fig. 21-18). In the first stage, at fairly low temperature, nitric acid is driven off and recovered. (Ruthenium can be expected to volatilize in this step.) In the second stage, at a temperature of at least 500°C, complete conversion of aluminum nitrate to aluminum oxide occurs. Fission-product nitrates which decompose at low temperature will also be converted to oxide form. The solid aluminum oxide is leached to remove soluble fission products which are then adsorbed on clay and fixed as described above. Unleached fission products remaining with the aluminum oxide are considered permanently fixed and are disposed of as a solid waste. The extent of leaching to be anticipated from the aluminum oxide matrix is given in Table 21-22 (Hatch, L. P., *et al.*, *Processing of High Level Atomic Wastes with the View to Ultimate Disposal AEC Rept. TID-7517*, pt. 1b, 1956).

A second system designed to convert the aluminum nitrate waste to aluminum oxide has been proposed which utilizes a fluidized-bed technique (Fig. 21-19) (Jonke, A. A., *A Fluidized-bed Technique for Treatment of Aqueous Nuclear Wastes by Calcination to Oxides, AEC Rept. TID-7517*, pt. 1b, 1956). In this

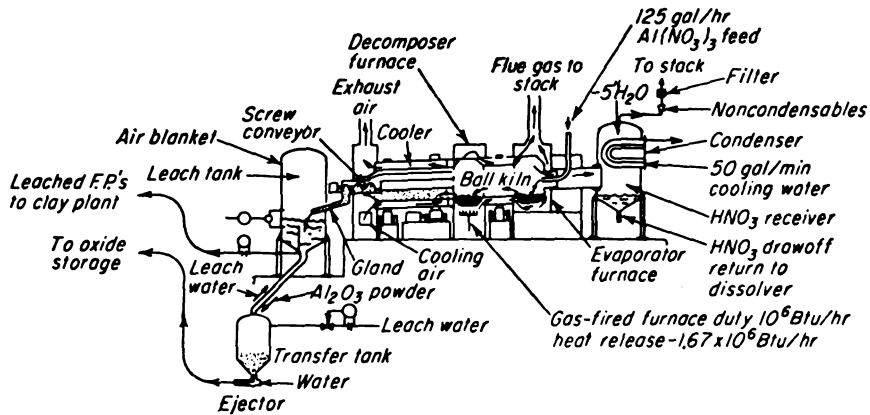


FIG. 21-18. Design of ball-mill kiln for concentrating fission products in Al_2O_3 produced from $\text{Al}(\text{NO}_3)_3$ content of waste. (Hatch, *Nucleonics*, vol. 13, p. 29, December, 1955.)

Table 21-22. Results of Leaching Tests on Fired Aluminum Oxide *

Firing temp, °C	Isotope	Leach time, days	Distilled water leach, % leached	Molarity of Al^{3+} in leach solution	Leach time, days	0.1N HNO_3 leach, % leached	Molarity of Al^{3+} in leach solution
350	Cs-137	5	80.0	0.058	5	86.0	0.111
350	Sr-90	5	61-65		5	70	0.07
350	Ce-144	27	81.4		52	80-90	0.36
350	Ru-106	27	3.5		52	7.0	0.464
650	Cs-137	4	90.5	0.0028	2	99.5	0.074
650	Sr-90	4	0.15-0.35	0.007	4	46-66	0.14
650	Ce-144	27	1.1		52	80-90	0.288
650	Ru-106	27	0.5		52	25-30	0.36
950	Cs-137	1	93.5	0.0014	1	100	0.002
950	Cs-137	17	98.5		32	100	0.072
950	Cs-137	18	89		33	89.95	0.062
950	Sr-90	1	0.22	0.004	1	25	0.009
950	Ce-144	27	0.05		52	25-30	0.074
950	Ru-106	27	1.5		52	7-8	0.058
950	MFP †	17	20.5		32	60	0.04
950	MFP	..			21	60 ‡	0.148
950	MFP	..			47	36 ¶	0.07

* Hatch, L. P., et al., Processing of High Level Atomic Wastes with the View to Ultimate Disposal, *AEC Rept. TID-7517*, pt. 1b, 1956.

† Mixed fission products.

‡ 6N acid used.

¶ .06N acid used.

system, a bed of aluminum oxide is maintained in a fluidized state by air passed upward through the bed. The bed is maintained at a temperature of 500°C by electric or gas heaters around the walls of the container or, if necessary, within the bed itself. The waste is injected into the bed through spray nozzles and is converted to the oxide on contact with the hot aluminum oxide particles. As the particles of oxide grow larger and settle toward the bottom of the calciner, they are withdrawn. Fine particles carried out by the air stream are removed by filters.

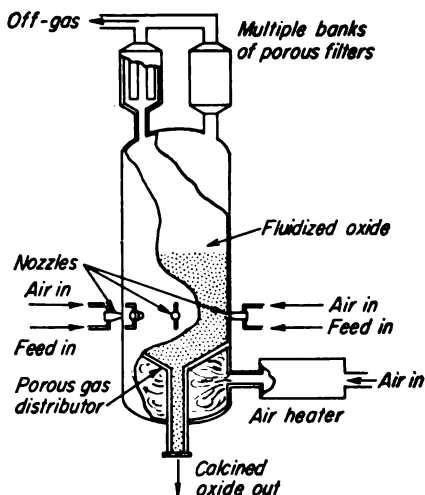


FIG. 21-19. Fluidized-bed calciner. (Jonke, A. A., *A Fluidized-bed Technique for Treatment of Aqueous Nuclear Wastes by Calcination to Oxides*, AEC Rept. TID-7517, pt. 1b, 1958.)

Zirconium-bearing wastes (from the dissolution of uranium-zirconium alloy fuel in hydrofluoric acid) can be treated successfully by calcination. Although zirconium fluoride does not decompose at the temperatures involved, it can be converted to the oxide by hydrolysis with steam at 700°C. A proposed flow diagram for this process is shown in Fig. 21-20 (Abriss, A., *et al.*, "Separation of Cesium and Strontium from Calcined Metal Oxides as a Process in Disposal of High Level Wastes," sponsored by American Institute of Chemical Engineers, in Session Waste Disposal II, Nuclear Engineering and Science Conference, Philadelphia, Pa., March, 1957).

Immobilization of fission-product wastes by incorporating the waste into ceramic masses is under investigation at the Oak Ridge National Laboratory. The waste solution is mixed directly with limestone, sodium carbonate, and shale to form a gel-like slurry which, upon heating, forms a sintered mass. Preliminary leaching studies indicate little leaching of strontium occurs (McVay, T. N., *et al.*, *Fixation of Radioactive Wastes in Clay-flux Micro*, AEC Rept. TID-7517, pt. 1b, 1956). Under proper conditions (high-specific-activity wastes, well-insulated container) it is anticipated that the mixture will sinter through the heat generated by decay alone. With lower-activity wastes, supplementary heat may be required.

A process for incorporating fission-product wastes in unsoluble silicate glass has been proposed (Durham, R. W., "Disposal of Fission Products in Glass," Paper 57-NESC-54, Nuclear Engineering and Science Conference, March, 1955, sponsored by American Institute of Chemical Engineers in Session Waste Disposal II). The particular waste used (7 N in nitric acid) was mixed with nepheline syenite, a naturally occurring glass-forming mineral, in the ratio of 1 g to 1 ml of waste. The resulting mixture forms a gel from which nitric acid can be recovered. The dried gel

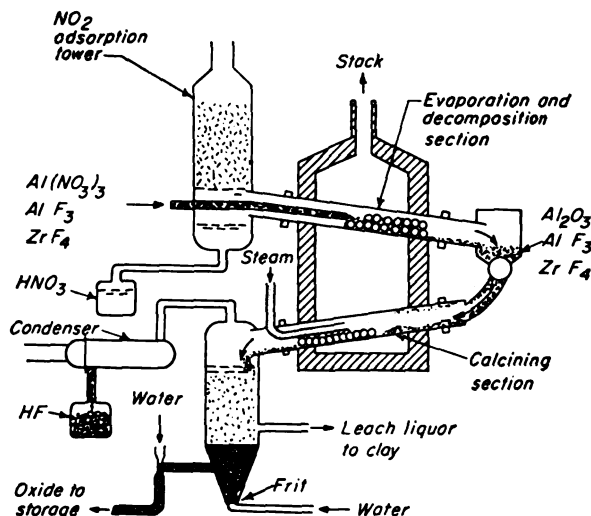


FIG. 21-20. Process flow diagram pretreatment of acidic fission-product waste. (Abriss, A., et al., "Separation of Cesium and Strontium from Calcined Metal Oxides as a Process in Disposal of High Level Wastes," Sponsored by American Institute of Chemical Engineers in Session Waste Disposal II, Nuclear Engineering and Science Conference, Philadelphia, Pa., March, 1957.)

is then heated to about 1300°C to form a fused mass. A glass with fewer occluded gas bubbles and with a somewhat lower fusion point can be obtained by neutralizing the acid waste with a fluxing agent such as $\text{Ca}(\text{OH})_2$ or NaOH prior to heating. Neutralization also decreases the loss of ruthenium through volatilization.

Leaching studies with Cs-137 indicated a daily leaching rate in the order of 5×10^{-6} per cent per day, depending to some extent on the exact composition of the glass. It was found that the use of calcium in the flux aided in incorporating aluminum into the melt.

Miscellaneous Methods of Disposal

Application of reactor waste to **useful purposes** is often considered as a method of defraying the cost of treating these wastes as well as being a method of holding the waste temporarily to permit some decay. Removal of the long-lived cesium and strontium isotopes from the bulk of the waste would permit disposal of the residue after a storage period of perhaps 10 to 15 years, compared with storage of

some 500 years if these elements are not removed. Such estimates, however, are based on the assumption that complete quantitative removal of Strontium-90 is effected. The effect of less than quantitative removal on the length of storage required (based on an installed reactor capacity of 2.2×10^6 mw, equivalent to 3 tons of fission products per day) is given in Table 21-23 (Rodger, W. A., Trends in Sepa-

Table 21-23. Effect of Residual Sr^{90} on Discardability of Waste *
(Basis: 2.2×10^6 mw installed reactor capacity = 3 tons fission products per day)

Sr remaining %	Decontami- nation factor	Years' storage required †
0	∞	13
0.00001	10^7	13
0.0001	10	90
0.001	10^5	180
0.01	10^4	270
0.1	10^3	360
1.0	10^2	450
10	10	540
100.0	1	630

* Rodger, *AEC Rept. TID-7517*, pt. 1a, 1956.

† To point where ~ 1 cu mile water will dilute activity to tolerance.

rations Process Development and Their Effects upon Waste Handling, *AEC Rept. TID-7517*, pt. 1a, 1956). Thus, while it may be economically feasible to extract a large percentage of the strontium and cesium for various purposes, it appears unlikely that the additional cost for complete removal will be justifiable. Use of bulk fission products as sources of heat may find some application.

Reduction in waste volumes through improved or new fuel-reprocessing methods would materially benefit waste-disposal operations. Two such methods which show considerable promise are based on the volatility of uranium hexafluoride and on the oxidative slagging of the fuel element in an oxygen-deficient atmosphere. In the first method, uranium is reacted with an interhalogen such as ClF_3 or BrF_3 and removed as gaseous UF_6 . The second method depends on preferential removal of stable fission-product oxides as a slag, and removal of cesium, strontium, and gaseous products by volatilization. Both methods provide only partial decontamination of the fuel, increasing the problems of subsequent refabrication, but result in very low volumes of waste.

SOLID WASTES

Sources

Solid radioactive wastes may result from the mining and processing of radioactive ores, from many laboratory activities, machine turnings from fuel fabrication plants and experimental projects, from chemical operations producing sludges and concentrates, and from contaminated equipment and trash in all types of operations.

The levels of activity associated with solid wastes may vary from a few times background to those requiring shielding or remote handling, with radiation levels of 2 to 150 r/hr at the container surface.

Disposal

Solid wastes may be disposed of by burial, by incineration if combustible paper, rags, or wood, or by remelting if metallic. Each method finds its use at installations operated by the AEC or its contractors.

Burial. The AEC operates burial areas for the disposal of contaminated solid waste and scrap (Morgan, J. M., Jr., Consideration in Evaluating a Burial Ground for Solid Wastes, *AEC Rept. TID-7517*, p. 249). Some are located as follows: Hanford Works, Washington, National Reactor Testing Station, Idaho; Los Alamos Scientific Laboratory, New Mexico; Oak Ridge National Laboratory, Tennessee; and the Savannah River Works, South Carolina.

These areas have been selected on the basis of several criteria aimed at reducing environmental hazards below tolerance levels, at the same time permitting constant surveillance and forbidding the use of the land by the public for generations to come. Burial is accomplished in trenches or cells, with covering by earth or concrete as needs dictate.

Physical factors which influence the selection of a burial site include topography, geology, soil characteristics, surface and ground-water hydrology, meteorology, and transportation facilities. No radioactivity can be permitted to escape to the environment, whether it be air, soil, or water. An ideal site would be in a remote section of the country, far from habitation and traffic by humans. For continuous operation in a safe manner over a period of 100 years, at least 1 sq mile of burial area should be reserved, with perhaps an equal fringe area surrounding it, and both areas enclosed with stout fences regularly patrolled. It would also be desirable to have any proposed burial site in a geographical area where solid radioactive wastes are likely to occur. This will limit transportation charges for rail and truck movement of the wastes.

Where solid wastes are produced and disposed of at the same installation, transportation is not the major factor. The criteria for selection of burial sites at the National Reactor Testing Station in Idaho have been presented by the AEC as follows (Anon., Solid Radioactive Waste Disposal at the National Reactor Testing Station, *AEC Rept. TID-7517*, p. 250):

1. Area not less than 10 acres.
2. Accessibility.
3. Not less than 15 and preferably 20 feet of unconsolidated sedimentary overburden on the bedrock.
4. Appreciable amounts of clay in the burial sediments.
5. Overburden sufficiently coherent to stand a short period in vertical or nearly vertical walls of burial trench.
6. Not directly up the ground water stream from existing or potential plant sites, or nearby populated areas.

The burial ground at the National Reactor Testing Station consists of 13 acres located 3 miles from the nearest reactor. The area is enclosed by a five-wire barbed

fence. Steel support posts are spaced at 10-ft intervals. Each post bears an identifying number which facilitates location inventory of material. The soil is a sand-gravel deposit exceeding 12 ft in depth over lava-reef formation. The water table is at a depth in excess of 300 ft in a lava-cinder formation.

Slit burial trenches are dug within the enclosure by means of a back hoe. An excavated trench is about 1,000 ft in length, 11 ft deep, and averages about 5 ft in width. Material is deposited nearly to the top of the trench, settling allowing several feet of cover. Trenches are periodically covered and compacted by the use of a bulldozer.

High-level wastes are deposited in the deepest portion of trenches, and shielding is accomplished by covering with low-level radioactive materials and finally with several feet of sand and gravel. In 3½ years four trenches were completely filled with approximately 3,800 cu yd of waste deposition.

Low-level routine radioactive wastes, less than 2 r/hr, are prepared within the operating areas for transfer to the burial site. This waste consists of low-level contaminated items such as absorbent paper, swabs, kleenex, various types of filters, bottles, and glass fragments. Normally, these are accumulated as follows:

1. In cardboard boxes 2 by 2 by 3 ft which are contained in the operating area in a steel shell with a hinged lid. When filled they are removed from the shell, taped tightly, and placed in the area pickup station.
2. In polyethylene bags which are contained in 32-gal garbage cans. When filled they are placed in the area pickup station.
3. Large items of low-level activity (planks, scrap metal, etc.) are usually loaded into a dump truck. Earth in a similar category is also handled in the same manner. Tarpaulin covering of the load is required prior to transfer when this method is employed.

Pickup is accomplished by means of a covered semivan, accompanied by a driver and two laborers on a twice-weekly schedule to all operating areas having waste for disposal. Following pickup and transfer to the burial ground all disposable containers (bags and boxes) are deposited in the trench under Health and Safety Division supervision. The location of burial is noted and placed in a burial-ground log for reference and inventory purposes.

High-level waste, measuring greater than 2 r/hr, is considered to be material which will occasion exposures to handling personnel of dosages approaching or exceeding daily exposure rates. The following transfer methods are employed at the National Reactor Testing Station for this type of material in order to keep exposures at a minimum:

1. Light materials up to 200 to 300 lb with surface area of approximately 4 sq ft or less on any face, producing radioactive emission up to 500 r/hr at any surface, are handled in a burial cart constructed for this purpose. The burial cart is a small remote-dumping vehicle equipped with a 20-ft towing handle, at the front end of which is located the dumping control. The trailer is towed by a pickup truck equipped with a bolt-type hitch. In order to dump the load into the burial trench the cart is easily disengaged from the truck and can be readily maneuvered into position and dumped by one man at a distance of 20 ft from the load.
2. Larger, heavier, and higher-level materials are generally transported by means of charging coffins provided by the contractor supplying the waste material. The

coffin is borne by a lowboy-type semitruck and tractor. Loading and unloading are accomplished by remote means, usually conventional pull-through or push-through procedure.

The most highly contaminated wastes are deposited in the lower levels of the trench, and in most cases shielding is accomplished by covering with subsequent loads of routine waste. However, in several instances immediate earth covering has been necessary in order to provide appropriately low radiation levels for personnel engaged in other burial assignments. At the present time all covered trenches are below 10 mr/hr at ground-surface level.

Costs for land burial may range from a few cents to a dollar a pound. Estimates are more readily made on the basis of volume of waste since bulk is the major problem with solid wastes.

Estimates of costs of solid-waste disposal by the previously described method for the National Reactor Testing Station are given in Table 21-24. These are published cost estimates for the calendar year 1955.

Table 21-24. Estimated Cost of Solid-radioactive-waste Burial, National Reactor Testing Station, 1955 *

Trench (excavation and backfill), 4,500 cu yd at \$1.20.....	\$ 5,400.00
Rocky flats waste (unloading), 37 at \$25.00.....	925.00
High-level loads, 42 at \$28.60.....	1,200.20
Dump-truck loads, 103 at \$27.00.....	2,781.00
Routine-truck loads, 107 at \$45.00.....	4,815.00
Total.....	\$15,121.20
Health and Safety and Security (time).....	\$ 1,083.75
Health and Safety and Security (transportation).....	173.40
Total.....	\$ 1,257.15
Cardboard boxes, 1,850 at \$0.75.....	\$ 1,387.50
Masking tape, 1,850 at \$0.23.....	425.50
Garbage cans (expended), 35 at 3.50.....	122.50
1,300 at \$0.27.....	351.00
Total.....	\$ 2,286.50
Total expenditure, \$18,664.85	
Total waste, 2,047 cu yd	
Unit cost of disposal (estimated), $\frac{\$18,664.85}{2,047} = \9.11 per cubic yard	

* AEC Rept. TID-7517, p. 255, 1956.

Costs at other AEC installations for on-site burial of solid wastes have been shown by Morgan (Morgan, J. M., Jr., Consideration in Evaluating a Burial Ground for Solid Wastes, AEC Rept. TID-7517, p. 249) to range from \$1.52 to \$9.40 per cubic yard, for volumes ranging from 500 to 14,400 cu yd/year.

Many contract operations take place at installations remote from the five national burial grounds. In the normal operation of the production facilities at Rocky Flats, Colo. (Ryan, E. S., and L. C. Farrell, *Costs Involved in the Off-site Disposal of Solid Wastes*, *AEC Rept. TID-7517*, p. 237), the Dow Chemical Company disposes of some low-level radioactive solid wastes by incineration or on-site burial. Many of the more radioactive wastes, both combustibles and noncombustibles, are transported 750 miles to the Rocky Mountains to the burial ground at the National Reactor Testing Station in Idaho because no suitable large-scale burial ground could be located near the plant site.

Before the Idaho burial ground was decided on, the possibility of placing the wastes in 30-gal drums was considered. These would have to be shipped to San Francisco, encased in concrete within a 55-gal drum, and transported to sea in accordance with the method discussed previously in this section. It was estimated that the cost of the operations at San Francisco alone would have been \$22 per 33-gal drum, or \$5.50 per cubic foot. Preparations at Rocky Flats and transportation to San Francisco are not included in the above figure. Shipment to Idaho was chosen as a far more reasonable alternative. Costs of this operation have been reported by the Dow Chemical Company and are summarized in Table 21-25.

Table 21-25. Cost for Off-site Disposal of Solid Wastes from Rocky Flats, Colo., Facility, April, 1954, to September, 1955 *

Number of trailers.....	41
Weight, lb.....	1,450,000
Volume, cu ft.....	41,900
Costs:	
Labor (Rocky Flats).....	\$ 3,305.00
Material (Rocky Flats).....	1,870.00
Freight.....	33,465.00
Disposal (Arco).....	4,090.00
Total costs.....	42,730.00
Cost/hundredweight.....	2.95
Cost/cu ft.....	1.02
Cost/ton-mile (750 miles).....	0.08

* Ryan, E. S., and L. C. Farrell, *Costs Involved in the Off-site Disposal of Solid Wastes*, *AEC Rept. TID-7517*, p. 241.

Incineration

Laboratory operations of many types lead to large quantities of **contaminants** such as cleaning tissue, absorbent materials, gloves, rags, wooden articles, and a host of other items. The logical step would be to reduce the volume of these solid wastes by combustion and dispose of the ashes and volatile radioactive substances separately. Incineration of combustible wastes has not been adopted on any large scale, but many experiments have been conducted to develop the procedure.

Harris and Weinstein of the New York Operations Office of the AEC carried out tests on open-field burning of low-level combustible wastes, in much the same way that burning dumps are used for municipal refuse (Harris, W. B., and M. S. Weinstein, Open Field Burning of Low-level Radioactive Contaminated Combustible Wastes, *AEC Rept. TID-7517*, p. 229). Care was taken to confine the wastes to prevent scatter of wind-blown unburned materials. A cage screen would suffice for this. For each of the three experiments increasing amounts of combustible radioactive material was burned in order to obtain data on contamination of the atmosphere by air-borne particulates and the ground by fall-out and ashes. Appropriate sampling stations were planned for each experiment for continuous sampling. Fall-out samples were collected on pressure-sensitive cellophane film 1 sq ft in area. These samples were ashed and counted for beta activity. Air samples were taken at the rate of 15 liters/min through 1½-in.-diameter Whatman No. 41 filter paper. These samples were counted for beta and alpha activity.

The results of this open-field-burning project are presented in Table 21-26.

Table 21-26. Fall-out and Air Contamination from Open-field Burning of Low-level Wastes *

Item	Case I	Case II	Case III
Weight, lb			
Original	500	4,000	13,500
Ash	170	290	1,320
Reduction, %	66	93	90
Volume, ft ³			
Original		200	770
Ash		22	75
Reduction, %		90	90
Activity, μ c			
Original	780	2,000	10,000
Ash	780	2,000	10,000
Estimated recovered fall-out	0.26	20	100
Fall-out, disintegrations/min/ft ²			
Max.	17	6,000	40,000
Min.	Negl.	12	46
Distance to background, ft.	400	400	800
Air contamination, μ c/ml ($\times 10^{-12}$):			
α	1.4		3.2
β	1.4	4.0	3.2

Case I: loose boxed material, volume indefinite.

Cases II and III: material packed by baling machine.

* Harris, W. B., and M. S. Weinstein, Open Field Burning of Low-level Radioactive Contaminated Combustible Wastes, *AEC Rept. TID-7517*, p. 229.

It may be deduced that no biologically significant contamination of the soil or air was created within a ½-mile radius by this type of incineration. Furthermore, wastes should not be permitted to accumulate to the quantity shown in case III.

Thus, if incineration of this type could be carried out frequently, as weather conditions permit, a laboratory remote from populated areas should be able to reduce the volume of its wastes in this manner. This would leave only the ashes to be disposed of by burial.

Incineration in fixed installations has been investigated from many angles and is applied to the disposal of combustible wastes at the Shippingport Reactor (LaPointe, J. R., and R. D. Brown, "Control of Radioactive Material at the Pressurized Water Reactor," paper sponsored by the American Society of Civil Engineers at Nuclear Engineering and Science Conference, Philadelphia, Pa., Mar. 12, 1957, Reprint N57-NESC-107, obtainable from ASME). The total cost has been estimated at \$0.30 per pound for 2,000 to 4,000 lb of waste/month. The flow sheet of the opera-

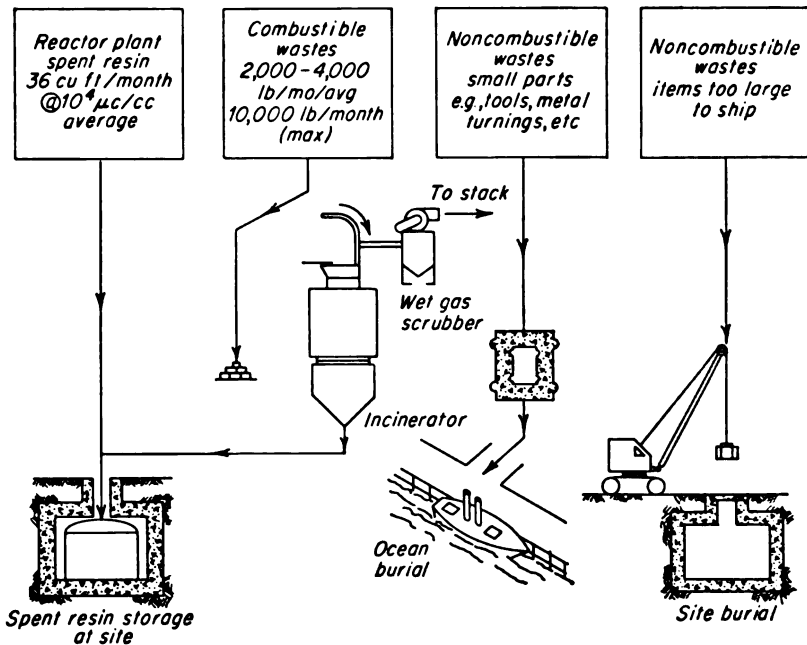


FIG. 21-21. Solid-waste-disposal process. (LaPointe, J. R., and R. D. Brown, "Control of Radioactive Material at the Pressurized Water Reactor," paper sponsored by the American Society of Civil Engineers at the Nuclear Engineering and Science Conference, Philadelphia, Pa., Mar. 12, 1957, Reprint N57-NESC-107, obtainable from ASME.)

tions is shown in Fig. 21-21. It will be noted that the flue gases are passed through a wet-gas scrubber. This removes most of the particulates greater than 5 microns and also cools the flue gas before filtration. The ash is dumped into an ash-slurry tank below the incinerator. The ash is then slurried with water and pumped into a storage tank. Contaminated ion-exchange resins are also stored into this tank until decay of radioactivity in incinerator ash and resins has proceeded to a safe level for burial.

Incineration of institutional wastes, such as those emanating from research in-

stitutions, hospitals, and others using radioisotopes, can include contaminated solids. Studies at the Johns Hopkins Hospital and other institutions in Baltimore (Krusé, C. W., P. V. Freese, *et al.*, Behavior of Institutional Incinerators when Used to Burn Radioactive Wastes, *AEC Rept.* NYO-4517, Nov. 1, 1952) showed that contaminated animals, cage cleanings, eggs, and other bulky combustibles having moderate levels of activity could be incinerated. Atmospheric contamination was shown to be a function of stack height. For a 10-m stack the radioactivity associated with I-131 would limit the charge to 100 $\mu\text{c}/\text{lb}$ refuse/hr. For a 50-m stack this value would increase to 1,125 μc . The ash-handling facilities, if manual and leading to local air contamination, would limit the refuse charge to 2 $\mu\text{c}/\text{lb}$ refuse/hr. For P-32 the limit would be 4 μc because of the ash hazard.

An incinerator using tangential overfire air has been developed by the U.S. Bureau of Mines (Corey, R. C., L. A. Spano, *et al.*, "Experimental Study of Effects of Tangential Overfire Air on the Incineration of Combustible Wastes," U.S. Bureau of Mines, September, 1953). Filtration will remove much of the particulates from the combustion gases, but the breathing hazards attending the removal of dry ashes or the contamination of water used in wet-collection methods are problems with all incinerators which are used to reduce the volume of contaminated wastes. The U.S. Bureau of Mines developed a method whereby all residues are fluxed in a molten compound as an integral step in the operation of the incinerator.

Sodium hydroxide is the best fluxing agent for ashes resulting from the incineration of wood, paper, cloth, or bone. Depending on the kind of ash being fluxed, sodium hydroxide will dissolve from 30 to 50 per cent of its weight of ash at 1000°F. Assuming a 10- to 15-fold reduction in the volume of 2,500 lb of combustible wastes, with 1 per cent ash, a 100-lb charge of sodium hydroxide will be sufficient to produce a molten flux which can be disposed of in steel containers. Thus, instead of the ash hazard limiting the charge of radioactive combustibles, contamination of the incinerator walls, the stack, or the atmosphere will become the limiting criterion of design and operation.

Mound Laboratory has reported on extensive pilot-plant work involving the burning of contaminated CWS filters, rubber, plastics, cloth, paper, and wood. It was found that (Schauer, P. J., Combustible Waste Disposal at Mound Laboratory, *AEC Rept.* MLM-549) all radioactivity in the off-gases was associated with particulate matter, with many particles in the submicron range. A decontamination factor for off-gas cleanup had to be on the order of 10^6 . This was obtained with a triple-stage wet-scrubbing system.

A commercial domestic-type gas-fired incinerator known as the Martin Burn-All, with a capacity of $1\frac{1}{2}$ bushels and a gas input of 20,000 Btu/hr, proved successful for the destruction of contaminated animal carcasses and associated biological and hospital wastes from the Medical School at the UCLA. The unit was installed outdoors (Silverman, L. B., and R. K. Dickey, Reduction of Combustible, Low-level Contaminated Wastes by Incineration, *AEC Rept.* UCLA-368, May 15, 1956) with a 12-ft stack, and some modifications were made. A special monitoring program was carried out to check the incinerator chambers and stack, as well as the ashes and off gases. Operation was supervised by health-physics personnel. Table 21-27 presents a summary of the loads placed on the unit in 2 years of operation within safe limits from all standpoints.

Table 21-27. Summary of All Loads and Activities Placed in Martin Burn-all Unit at UCLA, Jan. 13, 1954, to Dec. 1, 1955 *

Total weight of items burned	Major isotopes involved			
	Isotope	Total μ c	Max μ c/day	Avg μ c/burn day
3,400 lb of assorted combustibles including rabbits, rats, mice, excreta, peat moss, paper, etc.	I ¹³¹	20,000	500	70
	Sr ⁹⁰	3,000	200	10.5
	Cr ⁵¹	1,900	160	6.6
	Au ¹⁹⁸	1,000	150	3.5
286 days of this period were burn days when one or more loads were burned	S ³⁵	400	120	1.4
	Ca ⁴⁵	400	250	1.4
Avg weight of combustibles burned per burn day: 12 lb.	Y ⁹¹	130	30	0.45
	C ¹⁴	120	40 †	0.42
	MFP ‡	100	25	0.35
	Sum of others including:			
	Sr⁸⁹ Co⁶⁰ Fe⁵⁹ P³² }	50	1	0.17
	Totals	27,100		95

* Silverman, L. B., and R. K. Dickey, Reduction of Combustible, Low-level Contaminated Wastes by Incineration, *AEC Rept. UCLA-368*, May 15, 1956.

† Maximum concentration of C¹⁴ burned was less than 1 μ c C¹⁴ per 50 g of animal tissue plus required gas fuel.

‡ Mixed fission products.

Remelting of Contaminated Metals

Processing of natural uranium and other operations in the nuclear industry leads to great quantities of **contaminated iron and steel**. Not only does storage and possibly disposal by burial become an economic problem, but valuable scrap iron is removed from the channels of industry. The Health and Safety Laboratory of the New York Operations Office of the AEC conducted experiments on the effects of releasing mildly contaminated uranium-contaminated scrap to commercial channels (Blatz, H., J. H. Harley, and M. Eisenbud, Investigation of the Potential Hazard of Releasing Scrap Steel Contaminated with Uranium to Commercial Chemicals, *AEC Rept. NYO-1558*, June 15, 1951).

Analyses were made of steel and cast iron produced prior to 1940 in order to determine the natural uranium content of typical samples. Blatz (Blatz, H., Sale of Uranium Contaminated Steel Scrap Recommended, *Iron Age*, Sept. 25, 1952) prepared the data shown in Table 21-28.

Table 21-28. Analysis of Uncontaminated Steels *

<i>Sample</i>	<i>Uranium, $\mu\text{g/g}$ of steel</i>
Basic open-hearth.....	0.06, 0.1
Basic open-hearth.....	0.05, $\bar{x} < 0.01$
Bessemer.....	< 0.01 , 0.02
Bessemer.....	0.04, 0.2
Acid open-hearth.....	0.02, 0.02
Acid open-hearth.....	0.07
Lead-bearing steel.....	0.07, 0.02
Cast iron.....	0.6
W, Cr, V steel.....	0.02
Mean (steel only).....	0.0622

* Blatz, H., J. H. Harley, and M. Eisenbud, Sale of Uranium Contaminated Steel Scrap Recommended, *Iron Age*, Sept. 25, 1952.

Melting of uranium-contaminated steel in an induction furnace would produce a slag which could carry off a major portion of the uranium. To test this hypothesis five samples of grossly contaminated steel were melted after measurement for surface and total contamination. Original average contamination of the steel, as shown in Table 21-29, was 57.8 μg of U/g of steel. After melting in an induction, the melted steel was separated from the slag. The uranium content of the steel was re-

Table 21-29. Laboratory Melt, Contaminated Sheet Steel *

Sample	Alpha count † Avg each side	U analysis ‡		Alpha count ¶
		Before melting	After melting	
1	7, 14	117, 87	5.5	0.38
2	38, 9	92, 36	1.2	0.06
4	250, 65	71, 12.8	0.7	0.05
5	6, 0.4	141, 13.6	0.1	0.05
7	11, 20	1.6, 5.6	0.1	0.05
Mean.....		57.8	1.50	
Mean (omitting sample 1).....			0.47 (see Note)	
Scale.....		8,700		

Note: Sample 1 (after melting) was found to have small cavities in which there were inclusions of slag which could account for the higher uranium content.

* Blatz, H., J. H. Harley, and M. Eisenbud, Investigation of the Potential Hazard in Releasing Scrap Steel Contaminated with Uranium to Commercial Channels, *AEC Rept. NYO-1558*, p. 4, 1951.

† In units of 10 curies/m/100 curies/m/100 cm^2 .

‡ Micrograms U per gram steel.

¶ Counts per minute.

duced by a factor of about 40 to a level of $1.5 \mu\text{g/g}$ of steel. Sample 1, suspected of being high in slag, caused this higher value. Eliminating this, the mean was close to $0.5 \mu\text{g/g}$. These findings were later verified on a larger scale.

It was concluded that there would be no health hazard of the uranium in the slag, as even the highest uranium content of $6.6 \mu\text{g/g}$ would limit only 30 to 40 alpha particles and about 200 beta particles per square centimeter of surface per minute. Since the steel industry uses about $\frac{1}{4}$ ton of purchased scrap for every ton of finished steel produced, the contaminated steel would be considerably diluted. No hazard or injury to any commercial interests, except radioactivity-counting apparatus, would result from the use of such steel scrap after melting. Therefore, the AEC recommends the resale and remelting of uranium-contaminated steel scrap, thus reducing the storage and burial problem.

Air-dust samples revealed that no hazard should exist from dust or fumes during such remelting operations. With a single exception, all samples which were taken during the remelting processes indicated concentrations of uranium in the atmosphere below 10 disintegrations/min/cu m, or less than 1 per cent of the maximum permissible concentration. The one high sample showed a concentration of 80 alpha disintegrations/min/cu m. Since these samples were taken in the gas stream directly over the furnace, it can be assumed that even the remelting of large quantities could be safely accomplished.

The remelting process described above the heavy steel and iron can also be applied to uranium-contaminated sheet metal. Surface contamination between 1 and 10 mg of uranium per gram of metal sheeting ($\frac{1}{8}$ to $\frac{1}{4}$ in. thick) can be reduced to $1 \mu\text{g/g}$ by remelting. This applies to steel, stainless steel, nickel, and copper but not to aluminum. No hazard to personnel resulted. Contaminated slag can either be reprocessed for uranium recovery or dumped on normal slag dumps.

General References

Moeller, Significant Water Constituents of Reactor Coolants from the Standpoint of Induced Radioactivity, *AEC Rept. TID-7517*, 1956.

Love, S. K., and W. F. White, Analyses of Some Larger Streams of the United States at High and Low Flows, *Ind. Eng. Chem.*, vol. 48, p. 1248, 1956.

Culler, F. L., Physical Nature of High Level Radioactive Wastes, Notes on Fission Product Wastes from Proposed Power Reactors, *Oak Ridge Natl. Lab. Rept. CF 55-4-25*, 1955.

Culler, F. L., Chemical Nature of High Level Radioactive Wastes, Notes on Fission Product Wastes from Proposed Power Reactors, *Oak Ridge Natl. Lab. Rept. CF 55-4-25*, 1955.

Culler, F. L., Radiochemical Nature of High Level Radioactive Wastes, Notes on Fission Product Wastes from Proposed Power Reactors, *Oak Ridge Natl. Lab. Rept. CF 55-4-25*, 1955.

"Typical Annual Operating Costs Los Alamos Waste Treatment Plant (WD-2)," Sanitary Engineering Conference, Baltimore, Md., Apr. 15-16, 1954, *Wash-275*, p. 204.

Morton, R. J., and C. P. Straub, Removal of Radioactivity-chemical Coagulation (Ferrous Sulfate, Ferric Chloride, and Aluminum Sulfate at 1, 2, and 6 gpg), *J. Am. Water Works Assoc.*, vol. 48, no. 5, May, 1956.

Morton, R. J., and C. P. Straub, Coagulation and Settling Results—Jar Test Studies, *J. Am. Water Works Assoc.*, vol. 48, no. 5, May, 1956.

- Morton, R. J., and C. P. Straub, Strontium Removal by Chemical Treatment, *J. Am. Water Works Assoc.*, vol. 48, no. 5, May, 1956.
- Morton, R. J., and C. P. Straub, Removal of Radioisotopes by Phosphate Coagulation, *J. Am. Water Works Assoc.*, vol. 48, no. 5, May, 1956.
- Morton, R. J., and C. P. Straub, Laboratory Studies on Removal of Radioactivity from ORNL Process Wastes, *J. Am. Water Works Assoc.*, vol. 48, no. 5, May, 1956.
- Swope, H. Gladys, Ion Exchange Treatment of Radioactive Laboratory Wastes, *Proc. Ninth Ind. Waste Conf., Purdue Univ.*, p. 118, 1954.
- Brockett, T. W., Removal of Radioactive Materials by Conasauga Shale (Jar Stirring Method), *Proc. Eighth Ind. Waste Conf., Purdue Univ.*, 1953.
- Brockett, T. W., Removal of Radioactive Materials by Conasauga Shale, *Proc. Eighth Ind. Waste Conf., Purdue Univ.*, 1953.
- Brockett, T. W., Removal of Radioactive Materials by Various Clays, *Proc. Eighth Ind. Waste Conf., Purdue Univ.*, 1953.
- Glasson, T. J. E., et al., Costs of Evaporation at Knolls Atomic Power Laboratory (Based on Two-shift, Single-evaporator Operation), KAPL-594, 1951.
- Manowitz, B., et al., Operating Characteristics and Cost of Brookhaven Evaporator, and Vapor Compression Evaporation Handles Radioactive Waste Disposal, *Chem. Eng.*, vol. 62, pp. 3, 194-196, 1955.
- Rodger, W. A., et al., Evaporation Costs, Argonne National Laboratory, and Disposal of Radioactive Wastes at Argonne National Laboratory, *Proc. Eighth Ind. Waste Conf., Purdue Univ.*, pp. 474-491, 1953.
- Manowitz, B., R. H. Britton, and R. V. Harrigan, Relative Efficiency of Three De-entrainment Devices, Entrainment in Evaporators, *Chem. Eng. Progr.*, vol. 54, no. 7, pp. 313-319, 1955.
- Morgan, J. M., General Practices for Preparing Metal Containers for Sea Disposal, Wash-275, 1954.
- Rodger, W. A., 50-year Accumulation of Long-lived Isotopes and Required Dispersal Volumes, and Trends in Separations Process Development and Their Effects upon Waste Handling, *AEC Rept. TID-7517*, pt. 1a, 1956.
- Hatch, L. P., et al., Results of Leaching Tests on Fired Aluminum Oxide, and Processing of High Level Atomic Wastes with the View to Ultimate Disposal, *AEC Rept. TID-7517*, pt. 1b, 1956.
- Rodger, Effect of Residual Sr^{90} on Discardability of Waste, *AEC Rept. TID-7517*, pt. 1a, 1956.
- Estimated Cost of Solid Radioactive Waste Burial National Reactor Testing Station—1955, *AEC Rept. TID-7517*, p. 255, 1956.
- Ryan, E. S., and L. C. Farrell, Cost for Off-site Disposal of Solid Wastes from Rocky Flats, Colorado, Facility April 1954 to September 1955, and Costs Involved in the Off-site Disposal of Solid Wastes, *AEC Rept. TID-7517*, p. 241.
- Harris, W. B., and M. S. Weinstein, Fall-out and Air Contamination from Open Field Burning of Low Level Wastes, and Open Field Burning of Low-level Radioactive Contaminated Combustible Wastes, *AEC Rept. TID-7517*, p. 229.
- Silverman, L. B., and R. K. Dickey, Reduction of Combustible, Low-level Contaminated Wastes by Incineration, and Summary of All Loads and Activities Placed in Martin Burn-all Unit at U.C.L.A. Jan. 13, 1954 to Dec. 1, 1955, *AEC Rept. UCLA-368*, May 15, 1956.
- Blatz, H., J. H. Harley, and M. Eisenbud, Analysis of Uncontaminated Steels, and Sale of Uranium Contaminated Steel Scrap Recommended, *Iron Age*, Sept. 25, 1952.
- Blatz, H., Laboratory Melt, Contaminated Sheet Steel, and Investigation of the Potential Hazard in Releasing Scrap Steel Contaminated with Uranium to Commercial Channels, *AEC Rept. NYO-1558*, p. 4, 1951.

Hatch, L. P., and W. H. Regan, Jr., Processing of High-level Atomic Wastes with the View to Ultimate Disposal, *AEC Rept. TID-7517*, pt. 1b, 1956.

Gorman, A. E., Selection of Sites for Atomic Energy Plants, *J. San. Eng. Div., ASCE*, Paper 1168, February, 1957.

Moeller, D. W., Reactor Cooling Water—Theoretical Evaluation of Induced Radioactivities, *AEC Rept. TID-7517*, October, 1957.

Love, S. K., and W. F. White, Analyses of Some Larger Streams of the United States at High and Low Flows, *Ind. Eng. Chem.*, vol. 48, p. 1248, 1956.

Theis, C. V., Geologic and Hydrologic Factors in Ground Disposal of Waste, *AEC Rept. Wash-275*.

Terrill, J. G., D. W. Moeller, and S. C. Ingraham, Fission Products from Nuclear Reactors (A Quantitative Prospectus), *Proc. Am. Soc. Civil Engrs.*, Separate 643, p. 38, March, 1955.

Culler, F. L., Notes on Fission Product Wastes from Proposed Power Reactors, *Oak Ridge Natl. Lab. Rept. CF 55-4-25*, 1955.

Glueckauf, E., and T. V. Healy, "Chemical Processing of Fission Product Solutions," Paper P/415, vol. 9, pp. 635-639, International Conference on Peaceful Uses of Atomic Energy, Geneva, 1956.

Wash-275, Sanitary Engineering Conference, Baltimore, Md., Apr. 15-16, 1954, Office of Technical Services, Department of Commerce.

Cathers, G. I., Radiation Damage to Radiochemical Processing Reagents, *Proc. Intern. Conf. Peaceful Uses Atomic Energy, Geneva*, vol. 7, pp. 490-495, 1956.

Kunin, R., and R. J. Myers, "Ion Exchange Resins," chap. 2, p. 25, Wiley, New York, 1950.

Higgins, I. R., and R. G. Wymer, Diban—Ion Exchange Waste Disposal Scheme, I, *Oak Ridge Natl. Lab. Rept. ORNL-1984*, Nov. 28, 1955.

Kelley, W. P., "Cation Exchange in Soils," Reinhold, New York, 1948.

Browder, F. N., Liquid Waste Disposal at Oak Ridge National Laboratory, *Ind. Eng. Chem.*, vol. 43, pp. 7, 1502-1505, 1951.

Nicholson, E. L., Design and Initial Operation of the Radiochemical Waste Evaporator, *Oak Ridge Natl. Lab. Rept. ORNL-393*, 1949.

McCullough, G. E., Concentration of Radioactive Liquid Wastes by Evaporation, *Ind. Eng. Chem.*, vol. 43, pp. 7, 1505-1509, 1951.

Glasson, T. J. E., et al., Costs of Waste Processing by Means of Evaporation at the Knolls Atomic Power Laboratory, KAPL-594, 1951.

Manowitz, B., et al., Vapor Compression Evaporation Handles Radioactive Waste Disposal, *Chem. Eng.*, vol. 62, pp. 3, 194-196, 1955.

Manowitz, B., R. H. Britton, and R. V. Harrigan, Entrainment in Evaporators, *Chem. Eng. Progr.*, vol. 51, no. 7, pp. 313-319, 1955.

Talboys, A. P., Contamination of Plumbing by Low-level Radioisotope Wastes, *AEC Rept. NYO-4010*, May 1, 1952.

Anderson, C. R., et al., Design and Operation of High Level Waste Storage Facilities, *Proc. Intern. Conf. Peaceful Uses Atomic Energy, Geneva*, vol. 9, p. 640, 1956.

Coppinger, E. A., et al., Heat Problems in the Disposal of High-level Radioactive Wastes, *AEC Rept. TID-7517*, vol. 16, p. 475, 1956.

Wilson, E. E., Design Considerations of Storage Tanks for Radioactive Wastes, *Chem. Eng. Progr. Symposium Ser.*, vol. 52, no. 19, *Nuclear Eng.*, pt. IV, pp. 153-157, 1956.

Morgan, J. M., Technical and Economic Aspects of Disposal of Radioactive Waste at Sea, *AEC Rept. Wash-275*, 1954.

Brown, R. E., et al., Disposal of Liquid Wastes to the Ground, *Proc. Intern. Conf. Peaceful Uses Atomic Energy, Geneva*, vol. 9, pp. 669-675, 1956.

Clark, J. R., and H. A. Pohl, Earth Disposal of Radioactive Wastes at SRP, *AEC Rept. TID-7517*, pt. 1a, 1956.

Witkowski, E. J., Operating Experience in the Disposal of Radioactive Wastes in Open Pits, *AEC Rept. TID-7517*, pt. 1a, 1956.

Thurston, W. R., Summary of Princeton Conference on Disposal of High Level Radioactive Waste Products in Geologic Structures, *AEC Rept. TID-7517*, pt. 1a, 1956.

Hatch, L. P., *et al.*, Processes for High Level Radioactive Waste Treatment, *Proc. Intern. Conf. Peaceful Uses Atomic Energy, Geneva*, vol. 9, pp. 648-658, 1956.

Hatch, L. P., *et al.*, Processing of High Level Atomic Wastes with View to Ultimate Disposal, *AEC Rept. TID-7517*, pt. 1b, 1956.

Jonke, A. A., A Fluidized-bed Technique for Treatment of Aqueous Nuclear Wastes by Calcination to Oxides, *AEC Rept. TID-7517*, pt. 1b, 1956.

McVay, T. N., *et al.*, Fixation of Radioactive Wastes in Clay-flux Micro, *AEC Rept. TID-7517*, pt. 1b, 1956.

Durham, R. W., Disposal of Fission Products in Glass, Paper 57-NESC-54, Nuclear Engineering and Science Conference, March, 1955, sponsored by American Institute of Chemical Engineers in Session Waste Disposal II.

Rodger, W. A., Trends in Separations Process Development and Their Effects upon Waste Handling, *AEC Rept. TID-7517*, pt. 1a, 1956.

Morgan, J. M., Jr., Consideration in Evaluating a Burial Ground for Solid Wastes, *AEC Rept. TID-7517*, p. 249.

Anon., Solid Radioactive Waste Disposal at the National Reactor Testing Station, *AEC Rept. TID-7517*, p. 250.

Ryan, E. S., and L. S. Farrell, Costs Involved in the Off-site Disposal of Solid Wastes, *AEC Rept. TID-7517*, p. 237.

LaPointe, J. R., and R. D. Brown, "Control of Radioactive Material at the Pressurized Water Reactor," paper sponsored by American Society of Civil Engineers at Nuclear Engineering and Science Conference, Philadelphia, Pa., Mar. 12, 1957.

Corey, R. C., L. A. Spano, *et al.*, "Experimental Study of Effects of Tangential Over-fire Air on the Incineration of Combustible Wastes," U.S. Bureau of Mines, September, 1953.

Schauer, P. J., Combustible Waste Disposal at Mound Laboratory, *AEC Rept. MLM-549*.

Silverman, L. B., and R. K. Dickey, Reduction of Combustible, Low-level Contaminated Wastes by Incineration, *AEC Rept. UCLA-368*, May 15, 1956.

Blatz, H., J. H. Harley, and M. Eisenbud, Investigation of the Potential Hazard of Releasing Scrap Steel Contaminated with Uranium to Commercial Chemicals, *AEC Rept. NYO-1558*, June 15, 1951.

Blatz, H., Sale of Uranium Contaminated Steel Scrap Recommended, *Iron Age*, Sept. 25, 1952.

Section 22

CONTROL OF RADIOACTIVE AIR POLLUTION

By

LESLIE SILVERMAN, Sc.D., *Professor of Engineering in Environmental Hygiene,
Harvard School of Public Health, Department of Industrial Hygiene, Boston,
Massachusetts.*

Types of Contaminants	22-2
Sources of Air-borne Contamination	22-2
Methods of Control	22-5
Process Hoods	22-8
Special Laboratory Hoods	22-15
Glove Boxes	22-21
General Ventilation	22-23
General Principles of Cleaning for Radioactive Gases and Aerosols	22-25
Dispersion and Removal of Aerosols	22-32
Types of Equipment for Particulate Removal	22-34
Dry Filters	22-36
Wet Filters	22-41
Incineration and Evaporation	22-42
Performance and Choice of Equipment	22-43
Cleaning Gases from Power Reactors	22-43
Decay Products of Noble Gases	22-44
Associated Problems	22-45

CONTROL OF RADIOACTIVE AIR POLLUTION

Leslie Silverman

TYPES OF CONTAMINANTS

It is the purpose of this chapter to discuss briefly the **sources of radioactive contaminants**, to indicate their nature, and to describe the methods used to prevent their escape to workroom atmospheres and ambient environments related to nuclear-energy processes. Only those contaminants and aerosols which are radioactive will be discussed in detail; however, it is not intended to minimize the importance of toxic nonradioactive materials frequently encountered in the atomic-energy industry such as beryllium, boron, and fluorides.

Radioactive atmospheric contaminants can be divided into two general categories, gases and particulate matter. **Gases** are the natural formless state of substances evolved by evaporation or volatilization. **Particulates** are separated into two groups based on whether they are solid or liquid. In the solid form they can exist as dusts, fumes, or smokes, depending upon their composition or the nature of the process of generation. **Dusts** result from mechanical size reduction of solids and may be inorganic or organic ranging in size from submicroscopic to macroscopic. Usually 50 microns (μ) is considered an upper size limit for air-borne contaminants. **Fumes** result from combustion, sublimation, or condensation of inorganic solid materials and usually their size is less than 1 μ . **Smokes** are the product of organic-material combustion and usually have sizes below 0.5 μ .

Liquid particulates are usually known as mists, although **fogs** are used occasionally to denote excessive concentrations of submicron droplets. **Mists** are evolved by condensation, atomization, or entrainment of liquids by gases. Their size ranges from 0.1 to 25 μ . * When particulates are dispersed in air or gases, they are usually described as **aerosols**.

The particle size of typical radioactive and nonradioactive aerosols is shown in Table 22-1. The source of air contaminants is given below in relation to important nuclear-engineering operational conditions.

SOURCES OF AIR-BORNE CONTAMINATION

In **mining and ore processing (milling)** contamination arises from the mechanical dispersion or suspension in process air of uranium- or radium-bearing dusts. Radon

* Silverman, Leslie, *Industrial Air Sampling and Analysis*, *Ind. Hyg. Foundation Am. Chemistry and Toxicology Series Bull.* 1, Pittsburgh, Pa., 1947.

Table 22-1. Particle Size and Distribution of Some Inert and Radioactive Aerosols

Aerosol	Geometric * mean diam, M_g	Mass median * diam, M'_g	Standard geometric σ_g
Atmospheric dust from 14 U.S. cities avg (1).....	0.54 μ	0.97 μ	1.56
Atmospheric dust by electron micro- scope, Knolls Atomic Power Labo- ratory (2).....	0.028	1.12	3.05
Beryllium fluoride fume, BeF ₂ , from fur- nace pouring operation 10 ft from furnace (3).....	0.36	2.3	2.2
Iron oxide fume from open-hearth fur- nace (4):			
Before waste-heat boiler.....	0.047	0.65	2.55
After waste-heat boiler.....	0.057	0.82	2.60
Fission-product source pilot plant (5):			
Radioactive material using light microscope.....	0.47	2.25	2.17
Same using electron microscope..	0.014	0.14	2.39
Nonradioactive using light micro- scope.....	0.42	11.30	2.86
Uranium oxide fume produced by burn- ing scrap (6).....	0.12	8.11	3.29
Separations process stack effluent (dur- ing dissolving) (7) (size given by light microscopy, electron micros- copy shows a mean size of 0.05)...	0.2	3.1	2.6
Sodium oxide from burning Na metal (8). Note: lightfield microscopy gave a mean size at 900 $\times$ of 0.2 μ ...	0.04	0.17	2.0
Air-borne dust from incinerator ash dis- posal (9).....	0.43	101	3.86
Fume from ferrosilicon electric furnace during tapping (10).....	0.43	2.77	2.2

* Conversion to mass or geometric size based on mathematical conversion $\log M_g = \log M'_g - 6.91 \log 2\sigma_g$.

(1) Ives, J. E., et al., Atmospheric Pollution of American Cities for the Years 1931 to 1933, *Public Health Bull.* 224, GPO, Washington, D.C., 1936.

(2) Fitzgerald, J. J., and Detwiler, C. G., Collection Efficiency of Air Cleaning and Air Sampling Media, *Am. Ind. Hyg. Assoc. Quart.*, vol. 16, p. 123, 1955.

(3) Vorwald, A. J. (ed.), "Pneumoconiosis," p. 378, Hoeber, New York, 1950.

(4) Billings, C. E., Small, W. D., and Silverman, L., Pilot-plant Studies of Continuous Slag-wool Filter for Open-hearth Fume, *J. Air Pollution Control Assoc.*, vol. 5, p. 159, 1955.

(5) Fitzgerald, J. J., and Detwiler, C. G., Size Distribution of Particles Produced by Fission Product Source Pilot Plant, KAPL 1232, General Electric Co., Schenectady, N. Y., Nov. 11, 1954.

(6) Connors, E. W., Jr., and O'Neil, D. P., Efficiency Studies of a High Temperature Filter against Freshly Generated Uranium Oxide Fume, ANL 5453, Argonne National Laboratory, Lemont, Ill., June, 1954.

(7) Fitzgerald, J. J., Evaluation of KAPL Separations Process Stack Effluent, KAPL 1015, General Electric Co., Schenectady, N. Y., 1952.

(8) Linnatainen, R. C., and Mehan, W. J., Removal of Halogens, Carbon Dioxide and Aerosols from Air in a Spray Tower, ANL 5429, Argonne National Laboratory, Feb. 28, 1955.

(9) Megonnell, W. H., Ludwig, J. H., and Silverman, L., Dust Exposures during Ash Removal from Incinerators, *Arch. Ind. Health*, vol. 15, p. 215, 1957.

(10) Silverman, L., and Davidson, R. A., Electric Furnace Ferro-silicon Fume Collection, *J. Metals, Trans. AIME*, vol. 203, p. 1327, 1955.

gas is present associated with its parent radium. The normal ore-concentration procedures result in dusts from crushing, drying, and extraction processes.

The refining of crushed uranium ore results first in the orange oxide (UO_3) which in turn is reduced to the brown oxide (UO_2) from which the green salt (UF_4) is made. Contamination from these steps usually is particulates in normal or high-temperature gases. In stepwise processes * **uranium metal** (natural proportions of the uranium isotopes 99.3 per cent U-238, 0.7 per cent U-235) and **UF₆ gas** from which enriched or enhanced uranium metal is obtained are made from the green salt.† These steps, which involve forming, sintering, rolling, and other metallurgical treatments, can cause contamination ranging from fume to large dust particles. Both normal air and elevated gas temperatures may be involved. Corrosive gases are often encountered because fluorides and fluoride reduction are involved in many steps. The chief sources of contamination in and around **gaseous-diffusion plants** primarily involve exposures to gaseous fluorine and hydrogen fluoride. Uranium losses here largely involve scrap recovery.

Thorium is processed in a manner similar to uranium, and although the amounts are far less the exposures and contaminants created are similar in size and concentration. For general information it is pointed out that **beryllium** and **zirconium**, which are both very useful in reactor construction, also involve metallurgical processes involving fluorides or chlorides in producing the basic metal.

The production of **uranium metal**, **thorium metal**, and their alloys involves production procedures such as founding or casting, powder metallurgy, welding, milling, turning, drilling, finishing, and plating.

Both uranium and thorium metal particulates as oxides and fumes may be created by these processes. Contamination by alloys containing uranium, for example, with aluminum (Uralloy) and zirconium (Zircalloy) and others is possible. Many unique forms of metals and their oxides or hydrides may be encountered in such processes. Many transuranium elements including **plutonium** are handled which produce aerosols similar to uranium and thorium. These involve dissolving, precipitation, chemical extraction, and ion-exchange concentration methods.

Scrap materials from all machining operations are processed for uranium recovery. Such treatments also lead to possible air contamination by particulates.

The reduction of combustible wastes containing radioisotopes or uranium for disposal or recovery of material by **incineration** in controlled environments may result in air contaminants similar in size and properties to volatilization of the basic materials. Reduction of wastes containing iodine (I-131) or carbon (C-14) results in gaseous contamination, although usually particulates such as phosphates containing P-32 may be expected, or when uranium- or plutonium-contaminated wastes are treated their respective oxides will be produced. The particle size and loadings will depend on the elements and types of waste to be burned. Some values for incinerator aerosols are cited in Table 22-1.

Because of the small quantities of materials involved for large activity it is possible to conduct much radioisotope- or fissionable-materials handling in ordinary or

* Anon., In *Uranium Drama*, Chemical Process Plays Star Role, *Chem. Eng.*, vol. 62, p. 112, 1955.

† U.S. Atomic Energy Commission, "Atomic Energy Development 1947-1948," GPO, Washington, D.C., 1948.

special laboratory facilities. Since containment is a prime objective in all radioactive-materials handling it is necessary to conduct all such operations in enclosures such as **hoods, caves, and glove boxes** whenever possible. The types of processes involved include all those normally done in laboratory chemical approaches such as titration, electrochemical analysis, and spectrochemical identification. In addition, all those described above under fabrication and machining including plating and polishing can be conducted in such facilities. When activity levels are so high that personnel protection requires extensive shielding the operations are conducted in caves with remote-control manipulators, or "slaves."

The aerosols produced in the hood or enclosure will depend upon the processes involved, and their treatment as to cleaning and control will parallel that of the lesser-activity operations conducted with limited or no shielding.

The nuclear reactor or pile is the prime producer of heat, power, and radioisotopes as well as quantities of transuranic elements. In **air-cooled reactors**, when cooling air contains dust which is exposed to neutron flux near fuel elements or shielding, induced radioactivity results. Not only is the dust activated but active rare gases such as A-41 may be produced in quantity depending upon the flux. Other types of activity in the cooling-gas stream involve structural materials which are eroded to the atmosphere such as concrete, iron scale, and graphite. Another source of active uranium oxide particulates results from the **rupture or bursting of fuel elements** or slugs. These can cause the release of iodine as well as numerous fission products. The use of **spray-cooling water** containing minerals will also produce active aerosols. A last source to be mentioned is the decay of **radioactive gases in effluent cooling air** to particulates. Gases such as Kr-89, Kr-91, Xe-140, Xe-141, and it is believed all rare gases with half-lives greater than 3 sec, decay to active particulates. These produce fine (less than 0.2μ) aerosol particles.

METHODS OF CONTROL

The control of radioactive air-borne contaminants in the working environment may be handled by three principal procedures: **source control, local-exhaust ventilation, and general ventilation**. In practice these approaches may be applied alone or as a combination of two or more of the group.

Source control may utilize substitution, enclosure, or wet handling as a means of minimizing or reducing contamination of the atmosphere. Process-control methods similar to those used in reducing liquid-waste loss have been successful in reducing atmospheric contamination. Substituting **continuous-flow, or integrated-transfer techniques** in place of **batch operations** reduces the number of sources of contamination.

A typical example of process control has been used at the Hanford Engineering Works, Richland, Wash., where I-131 gases may be discharged to the atmosphere from dissolvers. Reduction of this discharge is accomplished by permitting longer periods of fuel storage and consequently longer periods of radioactive decay before the dissolving operation. Since I-131 has a tenth-life of approximately 80 days, an increase of storage time to 90 days or longer means a substantial reduction in the amounts of iodine evolved during the dissolving process. **Source control** may also utilize substitution, enclosure, or wet handling as a means of minimizing or elimi-

nating air or gas contamination. Many process-control methods in nuclear-energy work have been thoroughly studied with regard to substitution. In most instances the obvious alternate possibilities have been ruled out by the need for a specific material. However, in the case of application of nuclear materials to certain new industrial techniques such as tracers and instrumentation procedures, substantial gain may be made in this regard if a lower-toxicity or hazard-potential material can be selected.

Enclosures are almost paramount in regard to **containment of material** which might be disseminated to the atmosphere. Enclosures are necessary in many instances because inert atmospheres must be maintained and therefore contamination involves only the cleaning of inert gas to maintain the desired continuous operation within the system. In most situations whenever possible air volumes are reduced to an almost negligible amount by the use of adequate enclosures. The philosophy of reducing the amount of air or gas to be cleaned deserves serious consideration. The **cost of air cleaning** varies directly with the **air volume**. The degree of cleaning necessary depends on the type of contaminants involved, as fixed by the decontamination necessary.

Wet handling as a means of minimizing contamination is a procedure which also has merit and should be applied whenever feasible. In this procedure it is necessary to substitute wet-handling or wet-batching operations in place of dry-powder or dry-material transfer, and in some cases to introduce sprays in confined zones. It is possible, in many instances, to pass the surface-contaminated material through a liquid-seal transfer point.

In many instances the use of water or solvents is precluded because of the nature of the active materials. They may be hydrides or other compounds which react violently or cause rapid corrosion. Hence, wet handling if done must be with a liquid or material which cannot interfere with the process.

The principal method of control of radioactive materials at the sources has been the application of **local-exhaust ventilation**. The situations for control by local exhaust will depend upon the process. Those covered include process hoods for mining, crushing, and refining; metal fabrication and machining operations; special laboratory hoods, glove boxes, or ventilated enclosures; and special wet boxes for certain operations. The distinction between the type of process hoods, laboratory hoods, and ventilated enclosures whether dry or wet is largely on the basis of the particular operations involved. **Process hoods** designed for crushing, grinding, refining, and metal-producing operations, as well as those for machining operations, are not significantly different in principle from those used for other dust- and gas-producing operations in industry. Typical examples of such hoods are described in detail in references * on industrial ventilation.

In the case of handling more active materials, however, it is necessary to impose

* Hemeon, W. C. L., "Plant and Process Ventilation," The Industrial Press, New York, 1955.

"Industrial Ventilation. A Manual of Recommended Practice," 4th ed., American Conference of Governmental Industrial Hygienists, Committee on Industrial Ventilation, P.O. Box 453, Lansing, Mich., 1956.

Marks, L. S., "Mechanical Engineers' Handbook," 5th ed., McGraw-Hill, New York, 1951.

more severe requirements on the need for maintaining adequate control velocities at all times. This requires that special **air-flow-control devices** for laboratory hoods and glove boxes be provided. It is not intended to present here detailed designs for exhaust systems for radioactive materials, as the principles of local-exhaust hood design apply equally well to the handling of radioactive contaminants. Since differences in degree of contamination and permissible limits are several orders of magnitude lower, however, examples of the precautions necessary will be delineated.

In designing **exhaust hoods** or **enclosures** for radioactive air-contamination processes the following general concepts should be considered.

1. Reduce the cause of contamination as much as possible by modification or alteration by methods outlined above.
2. Place the hood as close to the source as possible without interfering with the operator's movements or with the process.
3. Shape and locate hood to take advantage of any dispersing or gravitational forces. In enclosure design this means make encompassing surfaces as small as possible but compatible with operation and process.
4. Provide a sufficient quantity of air to direct all air-borne particulates of hygienic significance toward the final exhaust opening.
5. The intrinsic value of the substances involved often requires that small chips and turnings be recovered. Settling chambers or screens for coolants are therefore necessary.
6. Many of the radioactive metals such as uranium and thorium have **pyrophoric** properties in nature, and when machining or grinding an adequate supply of coolant in the hood is necessary.
7. The **heavy metals** such as uranium have more inertial effects because of increased density, which increases duct settling as well as settling within enclosures. Provision to take advantage of this increased deposition should be done as close to the hood or enclosure as possible. Avoid long horizontal **duct runs**, but where necessary provide frequent cleanout doors.
8. The type of machine to be hooded obviously affects the physical shape and design of the enclosure. Visibility, flexibility, and ease of access and manipulation are essential items in the design. Provide **transparent-plastic construction** where possible to improve visibility.
9. In order to provide readily decontaminated surfaces where visibility is not paramount, it is suggested that nonpermeable materials be used whenever possible. If not, provide surfaces which have good decontamination characteristics as described elsewhere in this handbook.
10. Provide adequate **air entry to enclosures** at velocities which in themselves do not create disturbances or agitation of contamination within the hood or enclosure.
11. It is impossible to provide enough air velocity to control **large heavy metal particles** (20 μ and larger) directly by air flow; hence baffles or impingement surfaces should be placed in positions where such particles can be maintained within the hood or enclosure by their impact on the baffles.
12. In the case of **radioactive gases** generated within enclosures no special precautions are necessary, except to take care of their corrosive nature if any. Outside

of enclosures it is desirable to take advantage of any convective currents if they do not cause process or room contamination. The use of funnels or canopies over beakers, vessels, or vats often causes condensates to drain back into the original vented vessel and may cause contamination of this material from the funnel or canopy.

13. Where **cleaning devices** are located within or close to the hood or enclosure provide means of transferring these into the enclosure if they can become seriously contaminated with activity.

PROCESS HOODS

Some typical examples of hoods properly enclosed as described by Schulte, Hyatt, and Smith * are shown in Figs. 22-1 through 22-9. Similar enclosures are also shown for other machines by Taub.† These show hoods in use for machining operations at Los Alamos Scientific Laboratory. Figure 22-1 illustrates a typical **lathe**

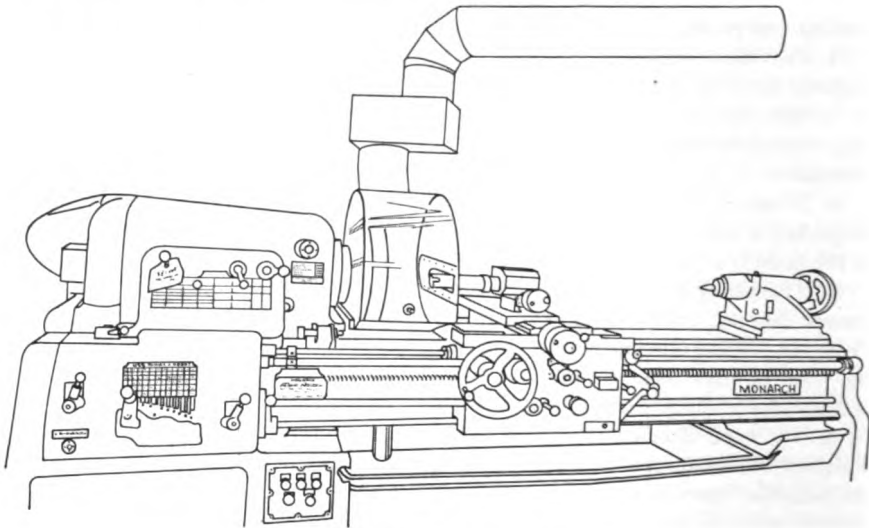


FIG. 22-1. Lathe hood.

hood used on enriched uranium. An inward velocity of 280 ft/min (85 m/min) is utilized. For heavy cutting the lathe hood door is closed. When machining to close tolerances or doing wet polishing, this velocity gives satisfactory control. In Fig. 22-1 just above the hood enclosure a chip trap consisting of a box provided with a screen may be seen. The duct velocity is high enough to entrain chips, and the trap separates them before they can enter the overhead exhaust duct.

* Schulte, H. F., E. C. Hyatt, and F. S. Smith, Jr., Exhaust Ventilation for Machine Tools Used on Materials of High Toxicity, *Arch. Ind. Hyg. Occupational Med.*, vol. 5, p. 21, 1952.

† Taub, J. M., Design Special Hoods for Machining Toxic Metals, *Iron Age*, vol. 179, p. 91, 1956.

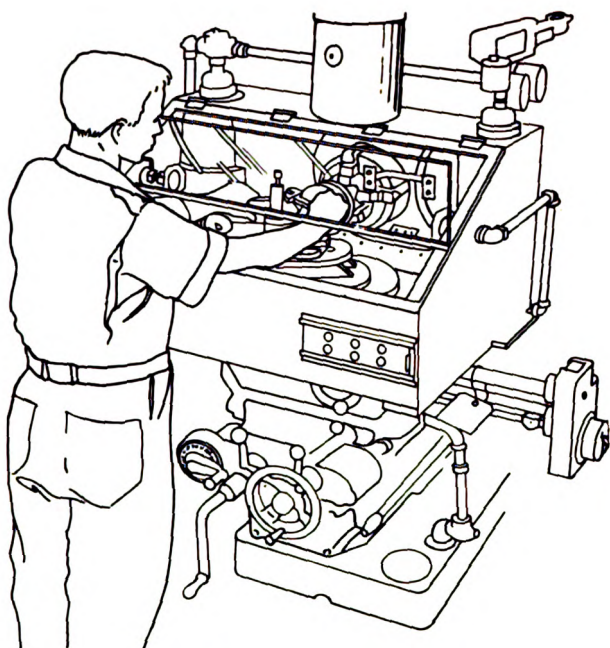


FIG. 22-2. Milling-machine hood.

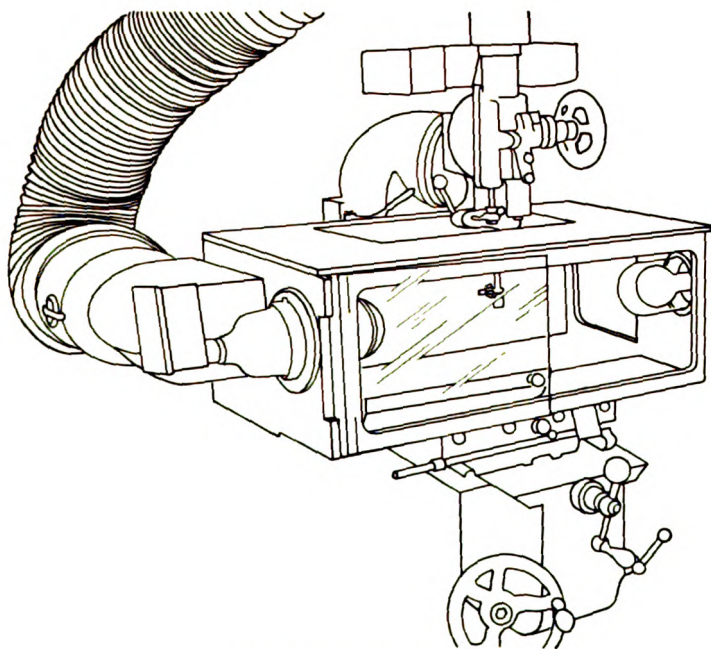


FIG. 22-3. Index-mill hood.

Figures 22-2 and 22-3 show **milling-machine enclosures**. When the door is open in the unit shown in Fig. 22-2, essentially the lathe velocity (275 ft/min, 84 m/min) is provided for control. The hinged door provides ready access. The index-mill hood shown in Fig. 22-3 does not require as high a velocity (200 ft/min, 61 m/min with the front open). The fixed opening at the top to allow for adjustments has an average velocity of 300 ft/min (91.5 m/min) when the hood is closed.

Figure 22-4 illustrates a relatively **inexpensive hood and enclosure**. Because the shaper is a device which ejects material with high-velocity trajectories, the enclosure is designed to stop chips within the space around the machine.

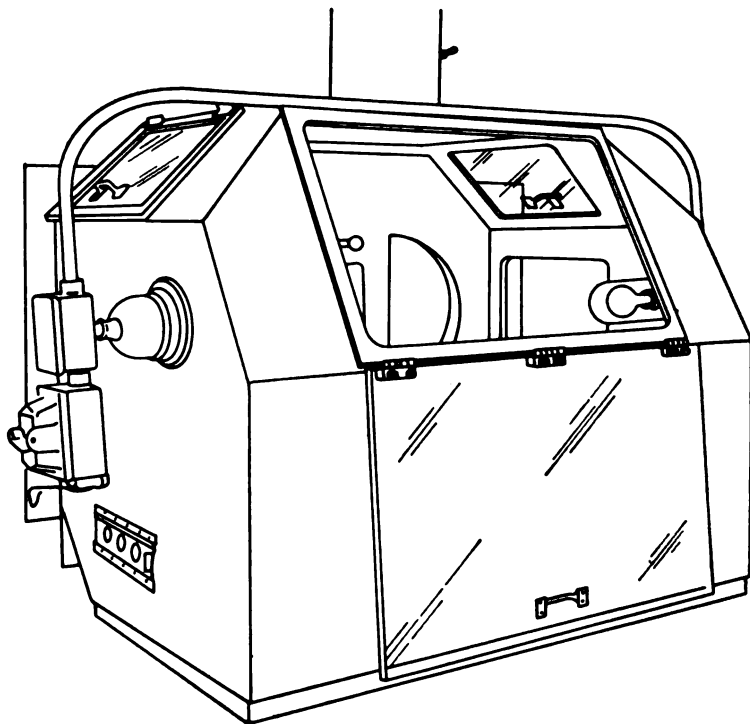


FIG. 22-4. Shaper hood.

Figure 22-5 illustrates a **power-hack-saw enclosure** in which it is possible to capture all the dispersed sawed metal dust by a small encompassing hood with an open top. Insertion of the flexible hose inlet as shown provides a satisfactory inward flow to the enclosure. Exhaust volumes of 600 cu ft/min will provide adequate control of the dust created during cutting. Because frequent access is desired, top ventilation seems the most suitable to apply.

Figure 22-6 indicates a small **handwork hood** for filing, sanding, and cleaning of small sections. It is essentially an open-slot box. A face velocity of 385 ft/min (117 m/min) is satisfactory.

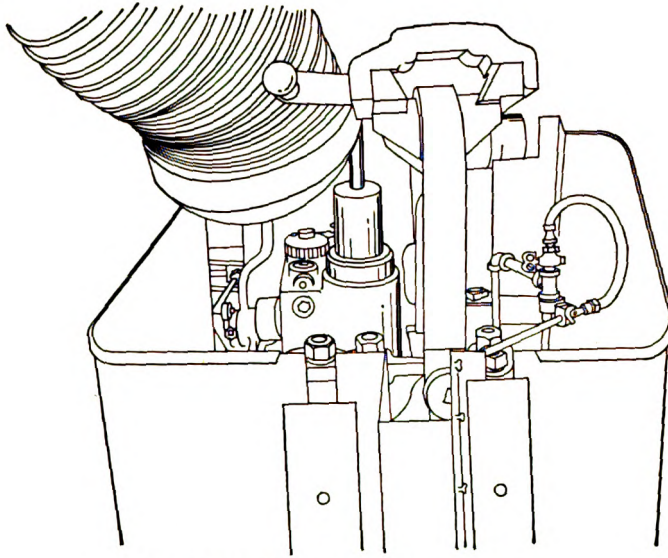


FIG. 22-5. Power-hack-saw enclosure.

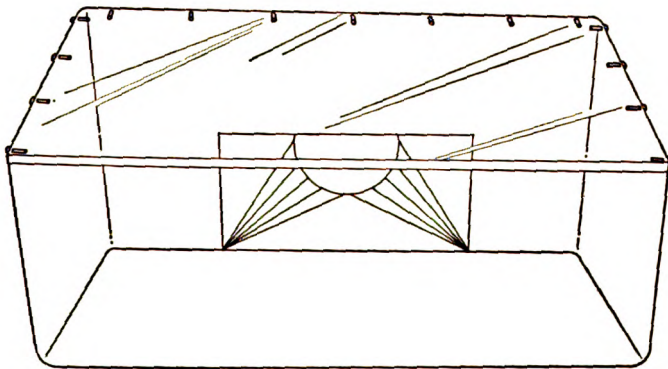


FIG. 22-6. Handwork hood.

Figure 22-7 illustrates a break-press hood where pieces of enriched uranium are broken prior to remelting. The breaker is enclosed in a small exhausted metal box inside the large exhaust hood enclosing the entire machine. When the small box door is open a face velocity of 500 ft/min (172 m/min) prevents dust loss. A face velocity of 550 ft/min (167.5 m/min) for the large opening is produced by the air exhausted from the small box.

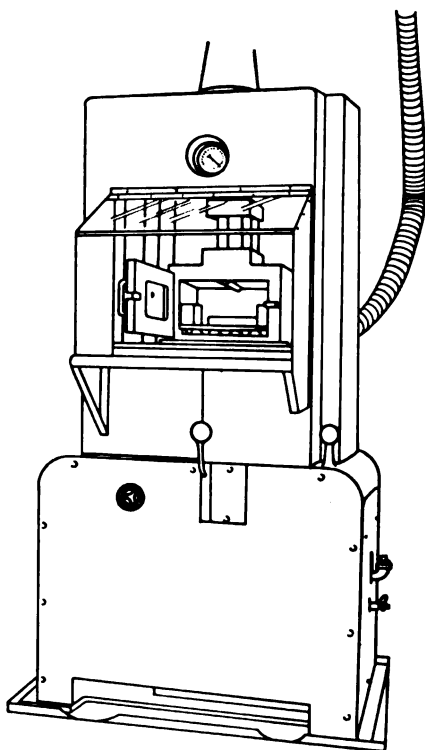


FIG. 22-7. Break-press hood.

Figures 22-8 and 22-9 refer to hoods used in beryllium machining operations. Figure 22-8 is a lathe hood which includes an enclosure and an internal 3-in. exhaust duct adjacent to the cutting tool. This duct is provided with an inlet screen to prevent chips from entering. The chuck is enclosed to prevent dispersion of dust. For efficient use the upper part of the large door must be closed. A face velocity of 250 ft/min (77 m/min) is necessary. The **grinder** shown in Fig. 22-9 is a surfacing machine. A small exhaust duct shown at the right is necessary to permit an airtight sliding fit for the moving bed. A baffle is provided to impact the coarse dust thrown off by the wheel. Velocities of 300 to 500 ft/min (91.5 to 152 m/min) through the hood openings are adequate for control.

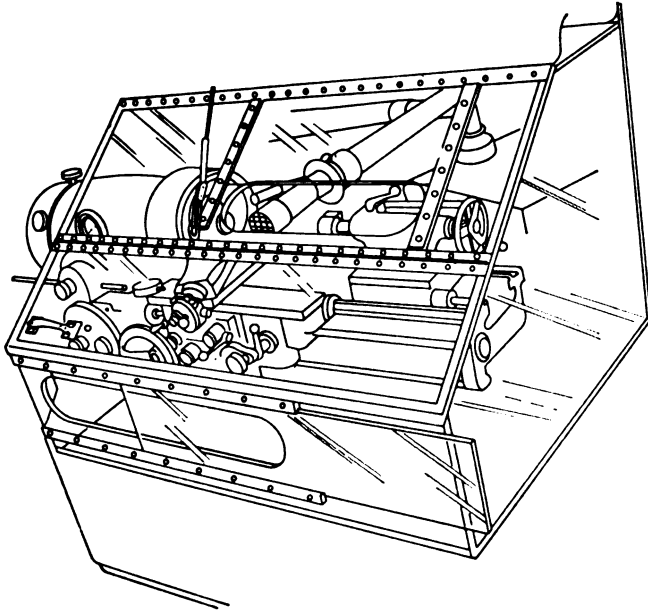


FIG. 22-8a. Lathe hood with internal exhaust duct.

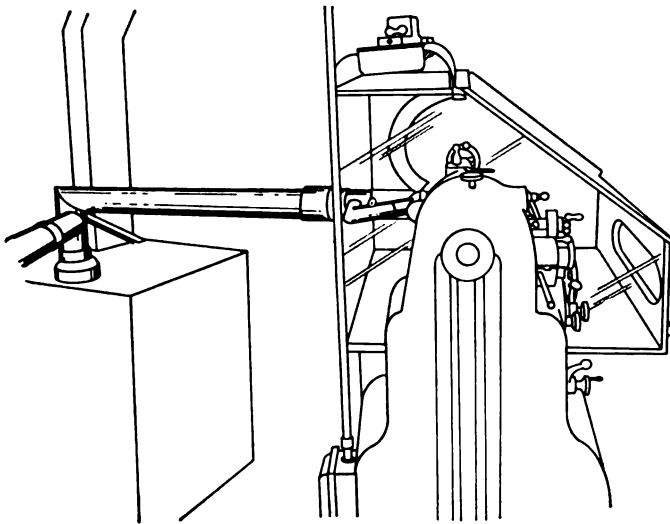


FIG. 22-8b. Lathe hood with internal exhaust duct (side view).

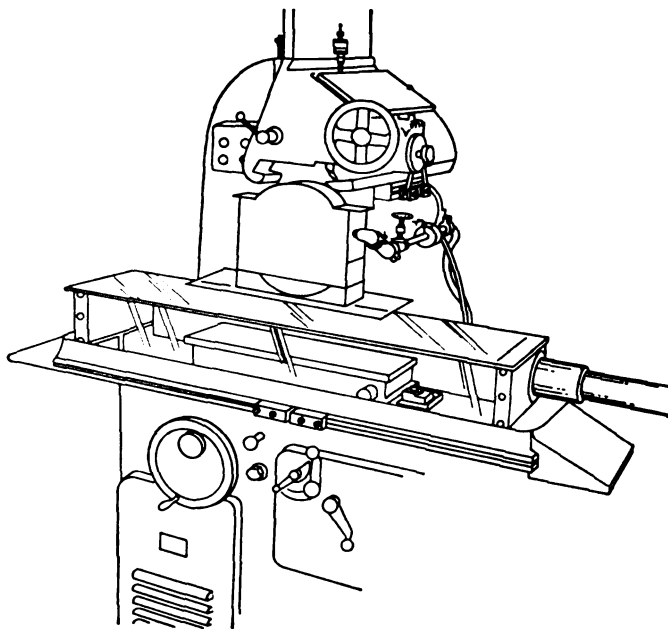


FIG. 22-9. Surface-grinder hood.

Table 22-2 summarizes the exhaust data for the above-described hoods, and Table 22-3 compares the air concentrations obtained during machining operations. These values are the result of prolonged sampling and represent average operating conditions.

Table 22-2. Ventilation Requirements for Machine Tools *

Machine tool	Metal machined	Air flow		Max working opening			
				Area		Velocity	
		cu ft/min	cu m/min	sq ft	sq m	ft/min	m/min
Lathe.....	Enriched uranium	500	152	1.8	0.17	280	85.3
Milling machine.....	Enriched uranium	390	119	1.4	0.13	275	83.9
Index mill.....	Enriched uranium	330	100.5	1.7	0.16	150-225	45.7-68.5
Shaper.....	Enriched uranium	540	164.6	2.7	0.25	200	61
Power hack saw.....	Enriched uranium	600	183				
Handwork hood.....	Enriched uranium	500	152	1.3	0.12	385	117.3
Break press.....	Enriched uranium	1,200	365.5	2.2	0.20	550	167.6
Lathe.....	Beryllium	200-250	61-76	0.75-2.5	0.73-0.23	100-270	30.5-82.3
Grinder.....	Beryllium	Variable	Variable	Variable	Variable	300-500	91.5-152
Lathe.....	Thorium	100	30.5	Not enclosed			

* After Schulte, H. F., E. C. Hyatt, and F. S. Smith, Jr., Exhaust Ventilation for Machine Tools Used on Materials of High Toxicity, *Arch. Ind. Hyg. Occupational Med.*, vol. 5, p. 21, 1952.

Table 22-3. Air Concentrations during Machining Operations *

Machine tool	Metal machined	Avg. concentrations, $\mu\text{g}/\text{cu m}$	
		Breathing zone	General air
Lathe.....	Enriched uranium	0.10	0.23
Milling machine.....	Enriched uranium	0.35	0.23
Index mill.....	Enriched uranium	0.16	0.23
Power hack saw.....	Enriched uranium	0.32	0.23
Break press.....	Enriched uranium		0.18
Lathe.....	Beryllium	0.49	0.26
Grinder.....	Beryllium	0.33	0.15
Lathe.....	Thorium	14.0	

* After Schulte, H. F., E. C. Hyatt, and F. S. Smith, Jr., Exhaust Ventilation for Machine Tools Used on Materials of High Toxicity, *Arch. Ind. Hyg. Occupational Med.*, vol. 5, p. 21, 1952.

SPECIAL LABORATORY HOODS

Because the quantities handled in the case of the more active materials have been quite small, ordinary **laboratory hoods** were used initially in nuclear development and have continued to be used, although they have undergone many modifica-

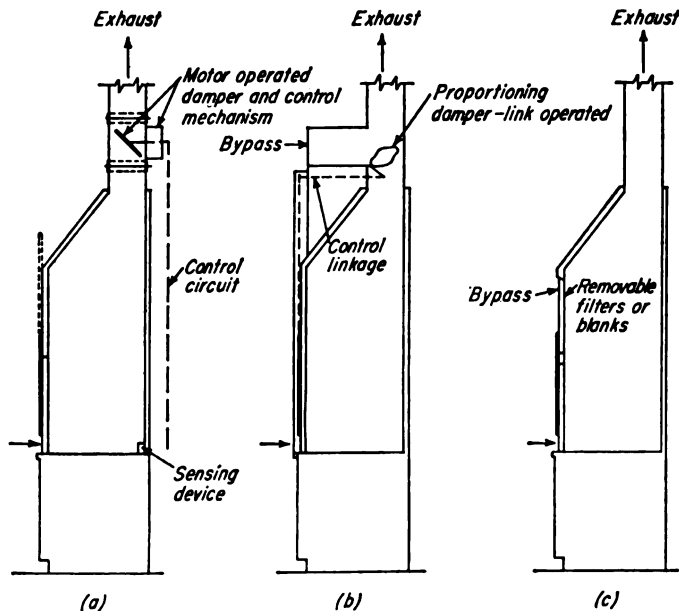


FIG. 22-10. Methods of controlling face velocity in radioactive-material hoods. (a) Controlled face velocities. (b) Proportional bypass link-operated. (c) Proportional bypass face opening. Entry loss = 0.25 VP (velocity pressure) plus dirty filter resistance. Duct velocity = 3,500 fpm min. Filters: (1) Prefilter. (2) Afterfilter. (*American Conference of Governmental Industrial Hygienists, "Industrial Ventilation Manual."*)

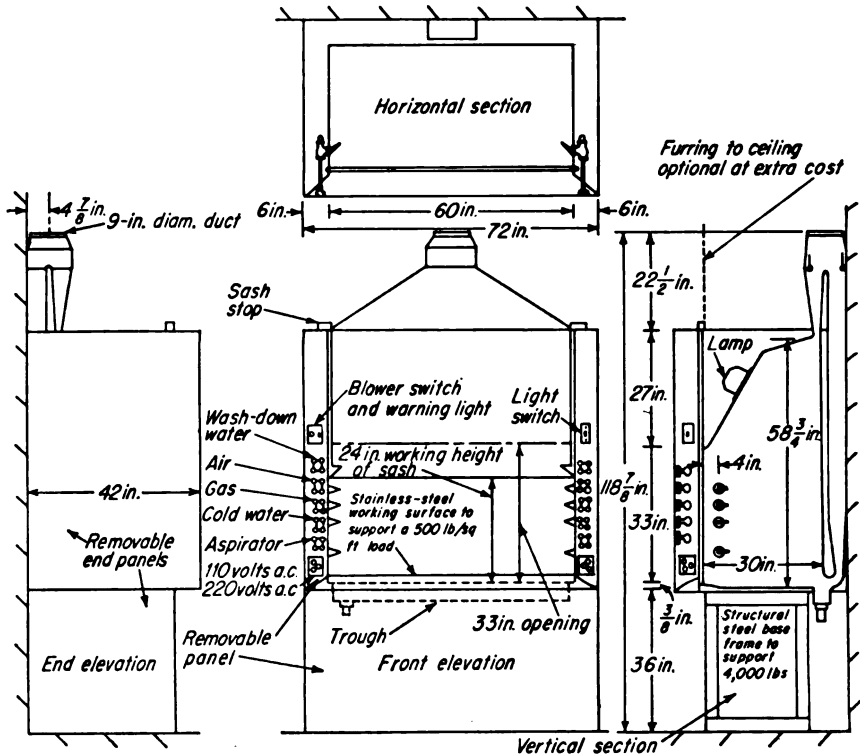


FIG. 22-11. Fume hood for handling radioactive isotopes. (Modified from design used by Oak Ridge Institute of Nuclear Studies.)

tions. The major refinement has been to provide constant air velocity at the opening, regardless of the position of the door or front sash. Such control has been obtained by mechanical linkages or geometrically shaped bypass dampers or openings, by pneumatic sensing elements, or by thermal sensing elements as shown in Fig. 22-10. Most of these techniques yield comparable results. Their selection is more often based on economics than on precision desired in control.

The basic need is to provide sufficient air velocity into the hood to prevent outward escape of contaminants. Loss of contaminants can be affected by abnormal heat loads within hoods, room air or cross drafts in rooms, and rapid movements (i.e., walking in front of hoods). Several satisfactory designs of hoods are commercially available.* Studies of their performance have been made by Kershaw,† and Schulte and his coworkers.‡

* York, J. E., *Ventilation and Air Conditioning for Laboratories, Heating and Ventilation*, vol. 50, p. 80, 1953.

† Kershaw, M. G., Panel Discussion, *Laboratory Design for Handling Radioactive Materials, BRAB Conf. Rept. 3*, Building Research Advisory Board, National Research Council, Washington, D.C., 1951.

‡ Schulte, H. F., E. C. Hyatt, H. S. Jordan, and R. N. Mitchell, *Evaluation of Laboratory Fume Hoods, Am. Ind. Hyg. Assoc. Quart.*, vol. 15, p. 3, 1954.

The use of **air foils** as a means of reducing entrance velocity to 50 ft/min is not generally satisfactory. They are effective under quiescent room conditions, usually obtained with supply air from a perforated ceiling. Rapid walking or movement in front of such low-velocity hoods, or high heat loads will disturb the control provided by low air velocities. Where large amounts of **fuming** or **spraying** are per-

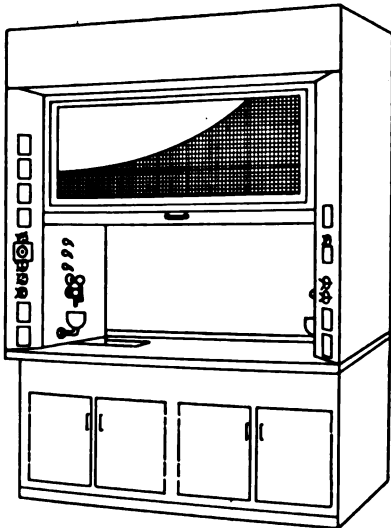


FIG. 22-12. Radioactive hood in which velocity is controlled by door opening. Grille above permits door to cause proportioning. (Modified from design used by Oak Ridge Institute of Nuclear Studies.)

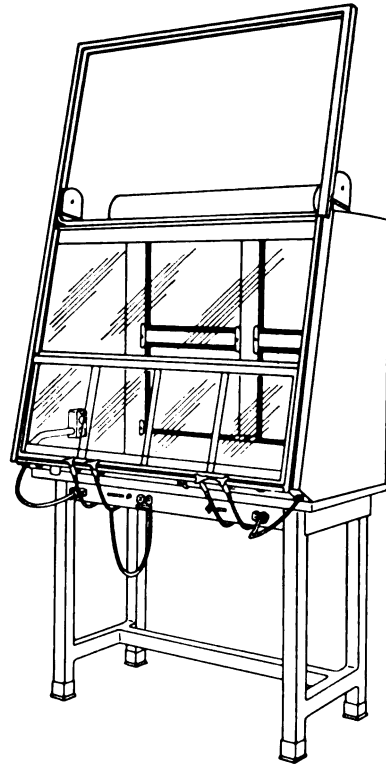


FIG. 22-13. Diagram of glassed-in hood which can be fitted with glove ports if desired.

formed in the hoods, velocities up to 180 ft/min will be desirable for control. In these cases the volume of air exhausted from the enclosure is more important than the face velocity. **Laboratory hoods** may be of the type shown in Figs. 22-11, 22-12, or 22-13. In the latter type, they may be readily converted to glove or dry-box operation.

In order to provide uniform distribution of flow at the face of the hood, several methods have been used. A **perforated wall** located at the back panel is an ideal approach and provides better distribution than a panel with slot entry at the top and bottom. A filter installed in the hood panel before the plenum is satisfactory in most instances. The plenum helps provide uniform distribution in the front of

hood. The chief advantages of the filter where applicable are its simplicity, ease of changing, and the fact that it maintains clean ductwork. Particle sizes as small as $1\ \mu$ can be removed by filters in this location at the back of the hood. Disadvantages are that they may decrease air flow as they become loaded or dirty. In some instances, filters can provide a radiation source when they contain large amounts of deposited active contamination. There are many locations where the prefilter can be placed adjacent to the hood, and as shown in Fig. 22-14, more than one position is possible.

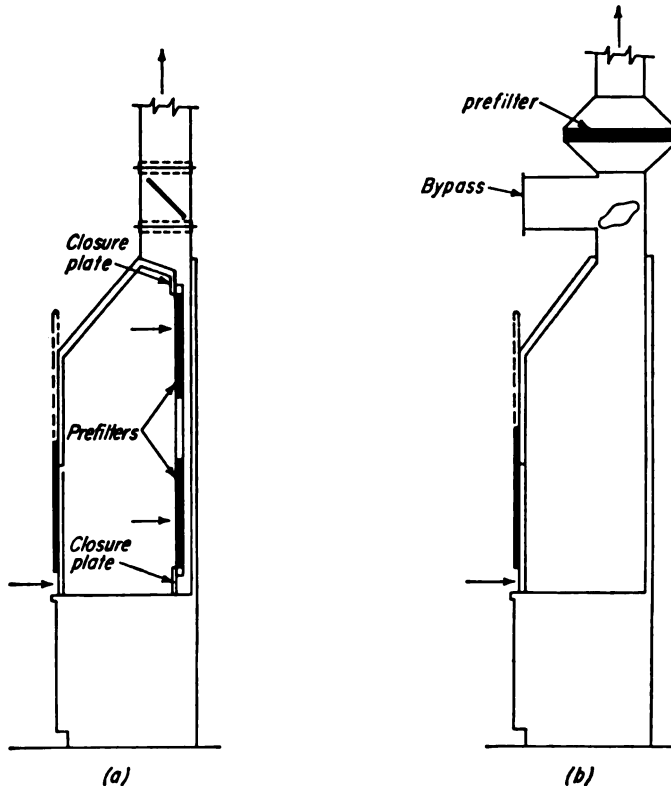


FIG. 22-14. Location of prefilters in radioactive hoods. (a) Prefilter in hood. (b) Prefilter in exhaust duct. Entry loss = 0.25 VP plus dirty filter resistance. Duct velocity = 3,500 fpm min. Filters: (1) Prefilter. (2) Afterfilter. (*American Conference of Governmental Industrial Hygienists, "Industrial Ventilation Manual."*)

Bypass air is introduced at either the floor or ceiling level, depending upon the design utilized by the laboratory. The hoods may be downdraft or updraft with the proportional bypass overhead or at floor level. Figures 22-15 and 22-16 show examples of both updraft and downdraft hoods. The advantage of the floor-sweep type is that it removes contamination at foot level. However, it draws in contaminated air at a level which is not prefiltered and which may add unnecessary inert-dust load to the final filtration systems.

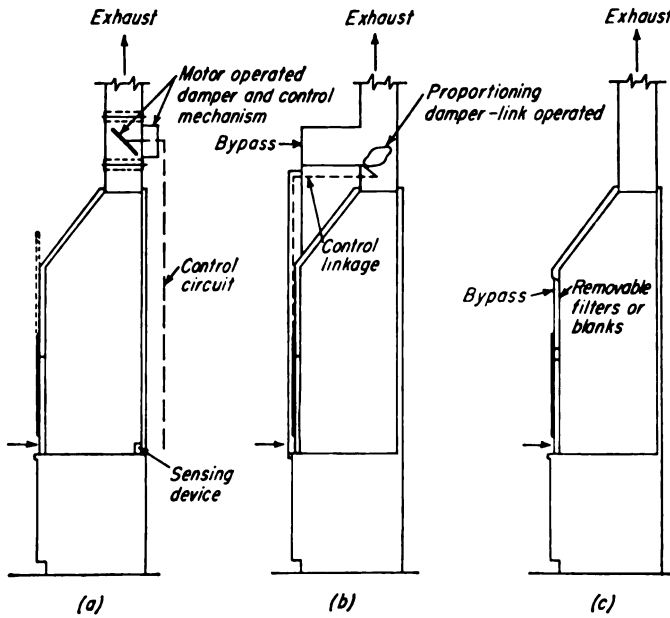


FIG. 22-15. Profiles of updraft hoods. (a) Controlled face velocities (100 fpm $\pm$ 20). (b) Proportional bypass link-operated (100 to 200 fpm). (c) Proportional bypass face opening (100 to 200 fpm).

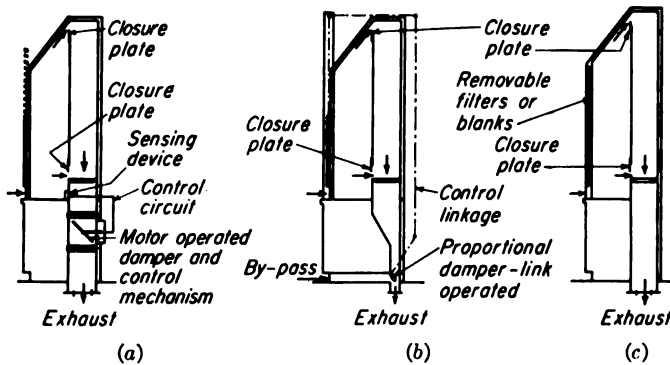


FIG. 22-16. Profiles of downdraft hoods. (a) Controlled face velocities (100 fpm $\pm$ 20). (b) Proportional bypass link-operated (100 to 200 fpm). (c) Proportional bypass face opening (100 to 200 fpm).

General conclusions in regard to the **design of laboratory hoods for active materials** are summarized below. Several of these observations were reported by Schulte and his coworkers.*

1. An average face velocity of 100 ft/min and a minimum velocity of 80 ft/min are necessary at any point on the face for hoods handling substances of moderate toxicity.

2. Hoods handling materials at high temperature or with extreme toxicity should incorporate an average face velocity of 135 ft/min. Face velocities should not exceed 300 ft/min.

3. The hood location in the room should be such as to minimize fire and explosion hazards to personnel. The hoods should be located as far away as possible from entrances, high-velocity air inlets, cross drafts, or other air disturbances.

4. Normal heat loadings will not adversely affect the performance of a hood with an average face velocity of 100 ft/min. Hoods with very high heat loads should have the major portion of the air exhausted through the top slot in addition to the increased velocity cited above.

5. The hood entrance should be kept free of obstructions and irregularities such as corner posts, sinks, and excessively depressed bases. A slight improvement in the air pattern is obtained by use of airfoils at all edges where application is feasible.

6. Hood doors do not contribute to the over-all performance of hoods but may be desirable for protection in case of a spill or an explosion. When doors are used, a servo-operated bypass of proper design is desirable to keep the face velocity below 300 ft/min when the sash is closed to a 6-in. opening and to provide a constant room-exhaust rate. Mechanical designs for uniform velocity-profile control are usually the most economic to install and are more trouble-free.

7. The most uniform face velocities are obtained by using filters or perforated panels at the back of the hood. In the simpler designs, or where filters or perforated panels are not desirable, a separate slot should be provided at the top and bottom.

8. Wherever possible, provision should be made for adjustment of the relative quantities of air drawn through the top and bottom slots. This adjustment is best done by means of dampers or levers outside the hood but not readily accessible to the user.

9. Exhausters should be located at the discharge end of the exhaust system to maintain a negative pressure in the conveying ducts.

10. Separate blowers are desirable in the case of special hoods or hood modules handling perchloric acid or agents such as reactive solvents or interhalogens.

11. Utility connections should be installed for maximum access for repair and maintenance but with minimum interference with air near the face of the hood.

12. Conservation of heated or conditioned air can be obtained only by minimizing the size of the hood opening rather than attempting to introduce unheated air at the hood face or reducing the face velocity.

* Schulte, H. F., E. C. Hyatt, H. S. Jordan, and R. N. Mitchell, Evaluation of Laboratory Fume Hoods, *Am. Ind. Hyg. Assoc. Quart.*, vol. 15, p. 3, 1954.

GLOVE BOXES

In accordance with the philosophy of containment and of maintaining balanced air supply and exhaust, the **glove- or dry-box** operation is most desirable for use with small amounts of materials. It is absolutely necessary to use this arrangement where inert atmospheres must be provided. The inert-atmosphere hood is described in detail in publications by Sherfey * and Gibb,† and will not be described extensively here since it is normally considered unique laboratory equipment.

A typical ventilated dry-box or glove hood design is shown in Fig. 22-17. It may be seen that the exhaust filter can be taken into the chamber and disposed of through a lock when necessary. Air supply to the hood is usually prefiltered by a small fiber-glass filter‡ or by means of edge filters made in small size of the type described by Silverman and First.§

Kelman, Wilkinson, Shuck, and Goertz § present in detail a large glove-box system for handling plutonium metallurgical operations. The filter and glove changing procedures are outlined in a useful manner. Details of the glove changing, which can be applied to all such boxes, are shown in Fig. 22-18. This procedure is essential in avoiding external contamination or dust permeation to the working environment.

Maintaining filters and cleaning devices in the exhaust system is inherent in all glove- or dry-box designs. Filters should be installed so as to be changed from

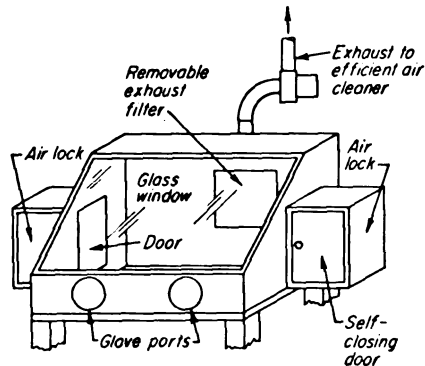


FIG. 22-17. Typical glove-box design for alpha- or beta-materials handling. $Q = 50$ cfm/sq ft of open-door area. Entry loss = 0.25 VP plus dirty filter resistance. Duct velocity = 3,500 fpm. Filters: (1) Inlet dust filters in doors. (2) Prefilter at exhaust connection to hood. (3) After-cleaner for final air cleaning. All facilities totally enclosed in hood. Exterior controls may be advisable. Arm-length rubber gloves are sealed to glove port rings. Strippable plastic on interior and air cleaner on exhaust outlet may be used to facilitate decontamination of the system. Filter units may be installed in the doors to allow the air flow necessary for burners, etc. (*American Conference of Governmental Industrial Hygienists, "Industrial Ventilation Manual."*)

* Sherfey, J. M., Inert Atmosphere Dry Box for Chemical Operations, *Ind. Eng. Chem.*, vol. 46, p. 435, 1954.

† Gibb, T. R. P., Jr., Inert Atmosphere Dry Box, *Anal. Chem.*, vol. 29, p. 584, 1957.

‡ Silverman, L., Cited in Kershaw, M. G., Panel Discussion, Laboratory Design for Handling Radioactive Materials, *BRAB Conf. Rept. 3*, Building Research Advisory Board, National Research Council, Washington, D.C., 1951.

§ Silverman, L., and M. W. First, Edge and Variable Compression Filters for Aerosols, *Ind. Eng. Chem.*, vol. 44, p. 2777, 1952.

§ Kelman, L. R., W. D. Wilkinson, A. B. Shuck, and R. C. Goertz, The Safe Handling of Radioactive Pyrophoric Materials, *Argonne Natl. Lab. Rept. ANL-5509*, December, 1955.

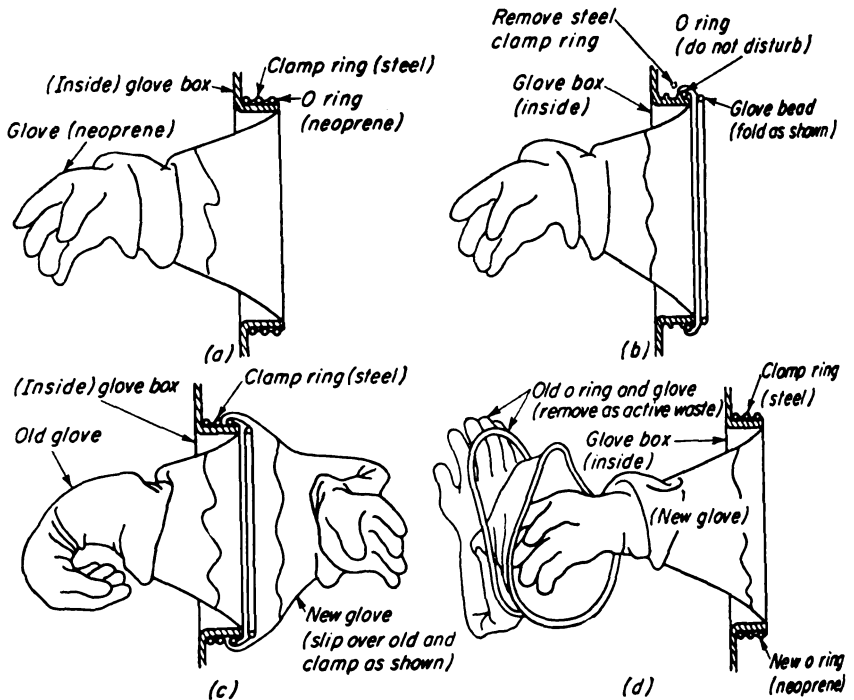


FIG. 22-18. Dry-box glove-replacement procedure. (a) Glove installed—ready for use. (b) Glove replacement—first step. (c) Glove replacement—second step. (d) Glove replacement—third step. (After Kelman, *et al.*)

within the box by glove manipulations. The type of filter to be selected depends on the decontamination necessary and will be discussed under Air Cleaning.

Certain types of glove boxes may require heavy shielding and simple manipulators. These are often called junior caves rather than strictly glove-box operations. Generally glove boxes are limited to low level or small amounts of alpha-emitting materials or those materials which have a high toxicity level and minimum or alpha radiation hazard.

The ventilation of **hot cells** is essentially the same as that for other types of enclosures. These cells may be operated with inert-gas or single-pass air systems. In all cases prefiltering is desirable to prevent outside dust from settling in the system, and external or exhaust filtering is essential. The filtering requirements may be much more severe in regard to construction and materials because of the gamma fluxes that may be present.

All the accepted air-cleaning filter change and installation procedures may be carried out within the hot cell, by use of remote slave manipulation as in normal chemical and physical operations within the cell. A typical ventilated hot cell is shown in Fig. 22-19. The same principles are applied to caves or heavy shielded operations (**junior caves**).

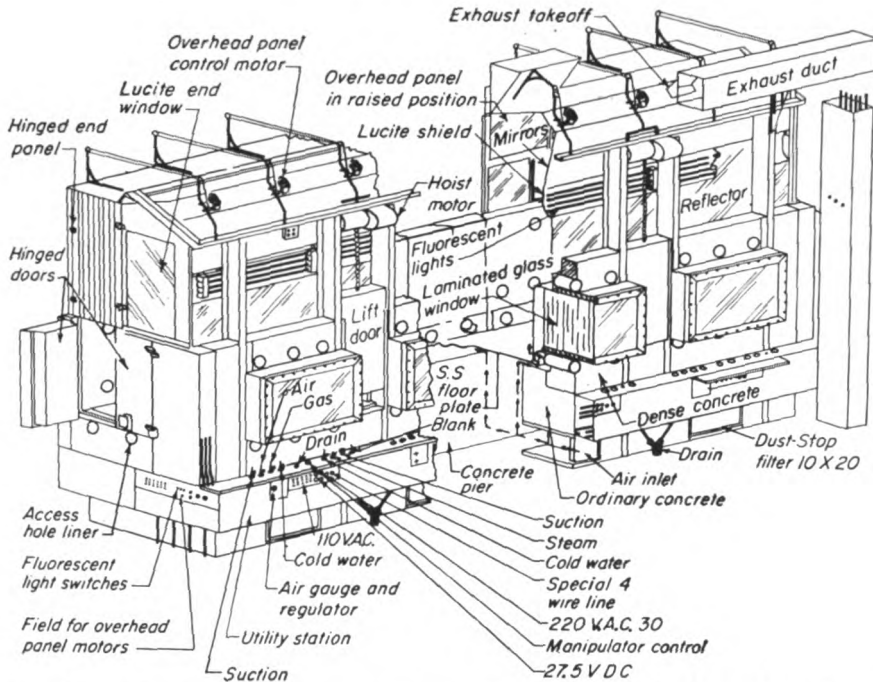


FIG. 22-19. Hot cell or cave for intermediate level (100 curies) work. Note exhaust ducts and fiber-glass prefilters. (After Fields and Youngquist, Argonne National Laboratory.)

GENERAL VENTILATION

General ventilation for atomic or nuclear-energy operations is different from customary applications in that the procedure for control depends upon the maintenance of a flow line of air movement in the direction of the sources of highest radioactive contamination. **The important factor to maintain is a movement of air from cold to warm to hot zones.** This may be illustrated by indicating that the supply air progresses from corridors to offices to laboratories or working areas and thence to the discharge.

In the case of refining and metallurgical production, the general supply ventilation is comparable with that for other metallurgical industries, inasmuch as the hazard is primarily one from particulates of chemical toxicity.

In using extremely active materials such as plutonium, it is necessary to provide that all contaminated air entering the laboratory will pass through the laboratory hoods, glove boxes, caves, or a filtered exhaust grille before discharge. Contamination from an operation which reaches the general room air will be largely removed by inlet filters to hoods or glove boxes or by the final filter system. Small amounts of contamination cannot be completely avoided, and in the case of any gas emissions, it is desirable to provide adequate dilution in the workroom air. General ventila-

tion is usually provided for normal comfort and air-conditioning requirements with very little excess to provide dilution for contaminants.

Where known volumes of contaminants are discharged to the room, the calculation of the additional amount of air to be supplied can be based on usual procedures, which involve assumption of distribution of air flow on the basis of the type of supply system. A perfectly satisfactory supply-air system for nuclear-materials handling in laboratories utilizes a perforated-panel ceiling. However, since this is quite expensive, trunk systems or multiple outlets have been substituted with satisfactory results. The use of recirculating types of diffusers should be avoided, as they deposit activity on the plates or vanes. Inlet louvers or elaborate draft diffusers should also be avoided, as they cause turbulence in certain sections of the workroom area and can cause deposition on the walls and ceilings of active particulate material.

In general, the air quantities for ventilation are based on dilution and distribution, and the usual techniques as given in references * will provide satisfactory results.

Features of **general ventilation for radioactive laboratories** which may differ from the usual case are summarized as follows:

1. Sufficient supply air must be provided to equal that exhausted by the hoods, glove boxes, or enclosures.

2. No recirculation of air shall be permitted in ventilating radioactive laboratories.

3. It is essential to maintain the laboratories in which active materials are handled at a slight negative pressure with respect to adjacent nonactive chemical or physical laboratories, corridors, or offices.

4. Supply air should always be taken from outside or from uncontaminated areas.

5. Adequate bypasses, cross-overs, and auxiliary equipment should be installed to provide for exhaust-air flow in the event of shutdown or power failures. It is customary to use reduced air flow (one-half or less) during nonworking periods, but positive flow should be available at all times.

6. In the case of highly active materials in sufficient quantities (curie levels) where ambient contamination might occur with power or exhauster failures, a parallel system for standby or emergency operation is essential.

7. Conveying velocities for exhaust ducts should range from 2,000 ft/min for gases to 5,000 ft/min for particulates of heavy metals. Such velocities should be high enough to prevent significant dust settlement which can result in local radiation sources.

8. Air- and gas-cleaning equipment should meet design decontamination factors and be readily accessible or removable without undue radiation exposures to maintenance personnel.

9. Air- and gas-cleaning devices should be provided with efficient monitoring if possible as well as continuous resistance or pressure-loss indicators or recorders.

* Hemeon, W. C. L., "Plant and Process Ventilation," The Industrial Press, New York, 1955.

Sherfey, J. M., Inert Atmosphere Dry Box for Chemical Operations, *Ind. Eng. Chem.*, vol. 46, p. 435, 1954.

GENERAL PRINCIPLES OF CLEANING FOR RADIOACTIVE GASES AND AEROSOLS

Air and gas cleaning for nuclear-energy processes differs from general air-cleaning problems in two respects. One is the high order of toxicity of the contaminants, and in the second the cleaning device often becomes dangerously contaminated and thus imposes health problems due to radiation or disposal of collected materials.

Radioactive gases and particulates in solid or liquid forms are the basic cleaning problems. Because of their high toxicity the degree of cleaning necessary is much more severe than with most nonradioactive materials. As discussed above, the air and gas cleaners, by their recovery function, provide solid or liquid contaminant materials which must be processed or handled. These range from rare gases such as Argon-41, which cannot readily be removed, to highly corrosive acid gases from metal-extraction processes, such as hydrogen fluoride. A wide range of particulate sizes are encountered as described previously. The finest aerosols are created by metallurgical fumes produced by burning or vaporization.

Requirements imposed by the low permissible concentrations specified for release to the atmosphere mean that efficiencies exceeding 99 per cent for aerosol particulates less than $1\ \mu$ are frequently necessary. For high-specific-activity materials of long half-lives, efficiencies that are a thousand times greater are often required. It is customary because of these exacting demands to talk of decontamination factors or penetration values ($1 - \text{efficiency}$) rather than efficiency. Decontamination factor is defined as $1/\text{penetration}$ or $1/(1 - E)$. For example, a factor of 10 means an efficiency of 90 per cent; similarly 1,000 means 99.9 per cent removal. For some processes values of 10^6 may be necessary for safe operation.

Three items are of prime importance in defining air- or gas-cleaner performance. These are (1) efficiency, (2) resistance to air or gas flow, and (3) life or operating period. The two most important in nuclear applications are the first and the last. The necessary power consumption to overcome the resistance to air flow cannot be disregarded but within reasonable limits can be provided without difficulty.

In commercial practice efficiency may be expressed (1) on a weight basis, (2) on a weight basis for certain particle-size ranges, (3) on a stain or discoloration or surface-area evaluation, or (4) by actual numbers of particles present. Since most radioactive contamination is on a mass basis, weight efficiency for certain particle-size ranges is a satisfactory method. Rapid testing of units with fine aerosols ($<0.5\ \mu$) may be done with light-obscuration procedures using a controlled-particle-sized aerosol such as **dioctylphthalate (DOP)** (diameter usually employed is $0.3\ \mu$).

Unit life in service is of great importance and is related to efficiency and resistance. Retention of particulates on filters influences flow resistance and life in relation to power consumption. Filter life becomes of special interest in minimizing exposure of workers required to change or clean components or units. For this reason, remotely controlled or constant-resistance methods are optimum for nuclear applications.

Radioactive gases impose difficult treatment methods for their removal. A major portion of the problem is created by rare gases such as Argon-41 and Krypton-85

or by Iodine-131. These gases are not easily removed by ordinary cleaning methods and are disposed of by different concepts, as discussed below.

The principal approach to their disposal from an economic standpoint, wherever permissible, is to utilize atmospheric dispersion and dilution from tall stacks in most cases. Other procedures that have been employed, depending upon the volumes involved, are containment by venting to a chamber or by pressurizing a chamber which is properly shielded and permitting adequate time for decay before atmospheric release.

Other methods involve scrubbing and adsorption at normal temperatures or at temperatures of liquid nitrogen or helium, adsorption on special materials, reaction with solid materials, and combustion and special reactions. Because of the economic factors involved, containment and decay are possible only with small gas volumes (less than 1,000 cu m). The use of solid adsorbents or freezing traps maintained at liquid-nitrogen temperatures is limited to gas flows of less than 10 cu m/min, as equipment and operating costs are in the range of several hundred dollars per cubic meter per minute of gas volume.

Scrubbing and absorption are utilized where gases and vapors are soluble or readily reactive with materials at controlled temperatures; good examples are acid and alkali-gas scrubbing or the reaction of iodine with silver or its nitrate. Iodine can also be adsorbed well in activated charcoal.

Stack dispersion or the use of **atmospheric dilution** has been employed for both gas and particulate dispersion by industry for decades. It is only in the last 25 years that the process has been studied intensively to ascertain all the meteorological and topographical factors involved. Obviously the method has been of great value as well as concern to atomic-energy operations. A fundamental reference is the recent AEC Handbook ("Meteorology and Atomic Energy," GPO, 1955), also *AEC Rept. AECU-3066*.^{*} This manual discusses all the meteorological variables involved in stack dispersion as well as current theories, formulas, and their experimental verification.

The use of a tall stack for dispersion from reactors or chemical-processing operation should always be a last resort or a safety factor beyond economically installed air or gas cleaners. When such cleaning is not economic, the cost of the stack and its associated meteorological control must be considered. Stacks of the size necessary for such operation are on the order of 200 to 300 ft (65 to 100 m). When constructed of masonry with adequate linings, their cost may be several hundred thousand dollars. **Meteorological control** associated with stack or plant operation has several **limitations**:

1. It is usually not economical.
2. Site location in regard to weather variations and topography may not necessarily be optimum.
3. Dispersion forecasts can be in error as much as 20 per cent of the operating time.
4. Changes in physical environment may result, such as adjacent plant buildings or communities.

^{*} "Meteorology and Atomic Energy," U.S. Atomic Energy Commission, GPO, July, 1955. (Also *AEC Rept. AECU-3066*).

5. Reactors or processes may not be flexible enough to provide instantaneous or undesirable prolonged shutdown without undesirable results.

In the case of stacks or other dispersion devices, meteorological control can offer **advantages** by providing assistance in the following ways:

1. Permit scheduling an auxiliary operation to take place during favorable weather conditions.

2. Justify the operation of special stack-gas treatment such as bypass air or gas cleaners which can be operated economically intermittently only; use of stack heaters for additional height during poor-atmospheric-diffusion conditions; or providing shutdown time for equipment repair, in which case the discharge can pass directly to the stack.

3. The use of ideal weather conditions for brief hazardous-gas releases.

There are certain basic **stack factors** which are associated with their effect on the operation or process. These are independent of dispersion problems. These are the gas parameters such as volume, temperature, pressure, and composition and the stack dimensional characteristics such as diameter, height, and surface roughness. For thermal-process operating conditions, the stack height is usually selected to provide adequate draft, if natural draft is employed, or auxiliary draft for forced- or induced-draft conditions. In the case of dispersion, stack height is selected on a basis of maximum ground-level concentrations at a known or selected distance from the stack. The important process variables for calculating stack dispersion are whether the process is continuous or discontinuous, volume or mass emitted per unit time, and temperature of discharge. The latter factor will affect the height of rise of the plume above the stack at low wind velocities and also the diffusion of the gases being discharged.

There are rather extensive mathematical theories and experimental verification of atmospheric diffusion which are presented in detail in the AEC Handbook mentioned above. Since space does not permit extended discussion here it will suffice to point out the practical aspects of these equations and to show their use.

The sources most commonly encountered in practice are as follows:

<i>Type</i>	<i>Application</i>
Continuous point source	Stack, vent, or elevated pipe discharge
Continuous line source	Array of any of those above
Instantaneous volume source	Explosion or batch operation producing clouds

The most frequently used case is the **continuous-point-source** equation from which is derived the maximum ground-level concentration and its location or distance from the stack. Plume width or height formulas have also been developed as well as extensions involving corrections for radioactive decay, ground-surface deposition, and diffusion in very stable atmospheres.

Sutton's formulas for a continuous point source are as follows:

Isotropic:

$$\chi(x_{1y}) = \frac{2Q}{\pi C^2 u x^{2-n}} \exp \left(-\frac{y^2 + h^2}{C^2 x^{2-n}} \right) \quad (22-1)$$

Anisotropic:

$$\chi(x, y) = \frac{2Q}{\pi C_x C_y \bar{u} x^{2-n}} \exp \left[-x^{n-2} \left(\frac{y^2}{C_y^2} + \frac{z^2}{C_z^2} \right) \right] \quad (22-2)$$

Q is doubled to allow for reflection from the ground; Q is the emission rate in grams per second, curies per second, etc.; χ = ground concentration (grams per cubic meter, curies per cubic meter, etc.) at a distance x (meters) downwind and y (meters) crosswind, from a source at height h (meters); $\bar{u}$ = mean wind velocity, meters per second; C_x, C_y, C_z = diffusion coefficients (meters) ^{$n/2$} in the x, y , and z planes, respectively; C = generalized diffusion coefficient (meters) ^{$n/2$} for isotropic turbulence; i.e., $C = C_x = C_y = C_z$; n = nondimensional parameter associated with stability; x, y, z = downwind, crosswind, and vertical coordinates from a ground point beneath continuous source meters.

The most useful forms of Sutton's equations are obtained by differentiation and maximizing Eq. (22-1) from which

$$\chi_{\max} = \frac{2Q}{e\pi\bar{u}h^2} = \frac{Q}{4.26\bar{u}h^2} \quad (22-3)$$

$$d_{\max} = \left(\frac{h^2}{C^2} \right)^{1/n} \quad (22-4)$$

where $d_{\max}$ is the distance downwind from the source. For the nonisotropic case:

$$\chi_{\max} = \frac{2Q}{e\pi\bar{u}h^2} \frac{C_x}{C_y} = \frac{Q}{4.26\bar{u}h^2} \frac{C}{C_y} \quad (22-5)$$

In order to use Eqs. (22-1) through (22-5) it is necessary to have some representative values for the exponent n and diffusion coefficients. Suggested values for the general coefficient C^2 as a function of stack height and stability are given by Sutton and are summarized in Table 22-4. These values are approximate for given localities

Table 22-4. Values of C^2 as a Function of Stack Height h and Stability Parameter n *

Condition	C^2 at various h values, m (m) ^a				
	n	$h = 25$	$h = 50$	$h = 75$	$h = 100$
Large lapse rate	0.20	0.043	0.030	0.024	0.015
Zero or small temperature gradient . .	0.25	0.014	0.010	0.008	0.005
Moderate inversion	0.33	0.006	0.004	0.003	0.002
Large inversion	0.50	0.004	0.003	0.002	0.001

* After Sutton.

and conditions. Actual measurements of C_x and C_y are desirable. Many values are available from local meteorological investigations.

Another frequently used expression for ground-level concentrations and one which is readily adopted to elevated sources is that due to Bosanquet and Pearson. *

$$\chi = \frac{Q}{(2\pi)^{1/2} p q \bar{u} x^2} \exp \left(\frac{-h}{px} - 2 \frac{y^2}{q^2 x^2} \right) \quad (22-6)$$

* Bosanquet, C. H., and J. L. Pearson, The Spread of Smoke and Gases from Chimneys, Disperse Systems in Gases, *Trans. Faraday Soc.*, vol. 32, p. 1249, 1936.

p and q are vertical and lateral diffusion coefficients. Equation (22-6) is for the special case of $n = 0$ since the maximum ground-level concentrations predicted by each differ only by a constant value. The Bosanquet and Pearson expression does not directly take into account the atmospheric stability.

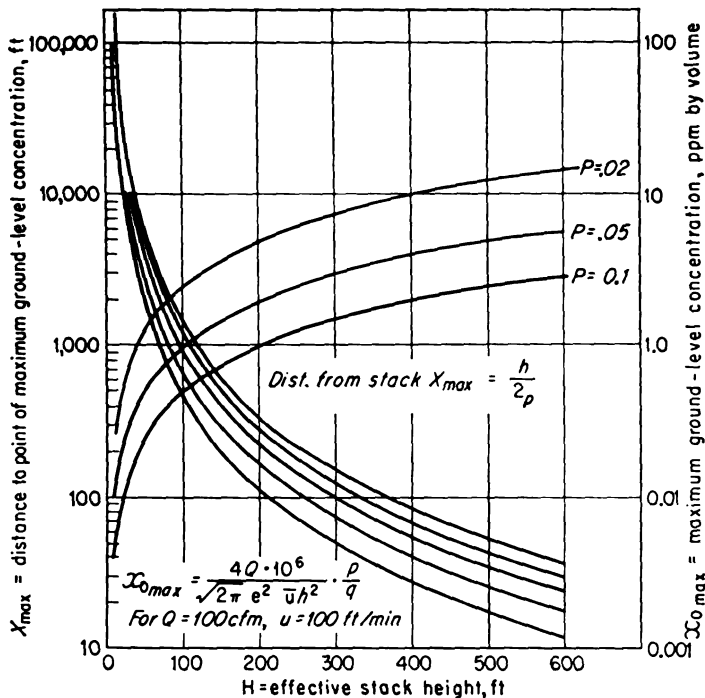


Fig. 22-20. Bosanquet and Pearson graphical solution for maximum ground-level concentration and distance from a stack. (After MCA "Air Pollution Abatement Manual.") Q = emission rate at atmospheric temperature (for gas), cu ft/sec. p = diffusion coefficient, dimensionless. u = mean wind speed, ft/sec. e = natural logarithmic base, 2.718. q = diffusion coefficient, dimensionless.

Turbulence parameters *	p	q	p/q
Low turbulence	0.02	0.04	0.50
Average turbulence	0.05	0.08	0.63
Moderate turbulence	0.10	0.16	0.63

* From "Air Pollution Abatement Manual," published by Manufacturing Chemists' Association.

Figure 22-20 presents a graphical solution of the maximum ground-level concentration obtained from maximizing Eq. (22-6). Conditions for a given Q and u are incorporated with curves for various turbulence parameters. Bodurtha * also

* Bodurtha, F. T., Jr., A Technique for the Rapid Solution of an Air Pollution Equation, *J. Air Pollution Control Assoc.*, vol. 5, p. 127, August, 1955.

gives a rapid technique for obtaining the values from the Bosanquet and Pearson expressions.

Falk *et al.** by fluorescent-particle studies have developed a method of using the above expression for predicting stack dispersion using simple measurements of wind speed and direction. They present a number of charts and modifications of the above formula for rapid calculations. Based on these data, the maximum ground-level concentration can be expressed as

$$\frac{\chi_{\max}}{Q} = \frac{0.19}{h^2 \bar{u}} \quad (22-7)$$

This is perhaps the simplest equation which can be applied for predicting purposes. The values of $\chi_{\max}$ were found to be within a value of 2 of the observed concentration about 85 per cent of the time and within a factor of 3 about 95 per cent of the time.

The equations and solutions outlined above usually apply to ideal conditions and do not take into account the **effect of surrounding structures and terrain** unless actual diffusion coefficients are measured. The use of scale models in low-velocity wind tunnels is one method of ascertaining the effects of surrounding structures, but actual field measurements may also be performed. Collins † describes a simple smoke-grenade technique for measuring flow around full-scale structures. In the absence of existing data a rule in practice is that *the stack height must be at least 2½ times above that of any surrounding buildings*. If this is not done the adjacent-building air flow may cause downwash and contamination at or near ground level near the stack or adjacent building.

In view of the variability and vagaries of the weather it is always desirable to be conservative in estimating stack heights to be provided for a new installation.

The essential feature of **containment and decay** for handling gases is their dependence upon small gas volumes of reasonably short half-life. The size and shielding of the storage vessel will depend upon the economics of the situation. Earth shielding may be adequate and some volume gain may be made by pressurizing; however, cost of compression equipment which is leakproof and maintenance-free is a major factor. As mentioned above, 1,000 cu m is the upper limit of maximum reasonable volume. If possible, cascading from smaller vessels is the desirable approach. The process is more easily adapted to intermittent processing rather than continuous handling. A containment system may also incorporate some reactant or catalyst if other than noble gases are involved. The philosophy of providing a containment vessel for a package or larger reactor, such as the experimental boiling-water reactor (EBWR), incorporates a chamber which provides a means of holding all gases and particulates in the event of a fire, explosion, or critical incident. In such cases it is possible to provide a bleed-off of air through a small standby gas cleaner which can handle the gases effectively over a long period at low flow rates.

* Falk, L. L., C. B. Cave, W. R. Chalker, J. A. Greene, and C. W. Thorngate, Development of a System for Predicting Dispersion from Stacks, *Air Repair*, vol. 4, p. 35, 1954.

† Collins, G. F., A Full-scale Study of Air Flow around Structures, *Ind. Wastes*, vol. 2, p. 109, July-August, 1957.

The usual techniques of **gas absorption and adsorption** for readily soluble or reactive gases may be employed for radioactive gases other than the rare gases. These include solid-gas reactions such as recovering hydrofluoric acid on marble chips or limestone pellets or adsorption of iodine or sulfur dioxide on activated charcoal. Recoveries in these cases can be within the range predicted for air-pollution control or chemical-engineering problems, and it is not intended to discuss them here, as they are adequately covered in references such as the "Air Pollution Handbook" by Magill, Holden, and Ackley.* Devices designed for gas absorption with reactive gases should be optimized for radioactive-gas absorption; i.e., maximum retention time should be allowed, especially where the diffusional process controls the rate of absorption.

In the case of iodine removal, for example, it has been found that alkali-spray scrubbers will remove from 85 to 95 per cent but to obtain higher removal requires the use of silver reductors. A silver-reductor unit consists of a tower packed with absorption surfaces such as Berl saddles or Raschig rings coated with metallic silver or silver nitrate. These are maintained at temperatures of 150 to 175°C by steam-heated jackets or electric heaters. They should be preceded by filters to prevent fouling, plugging, or deposition on the adsorptive and reactive surfaces. Packed fiber-glass filters are utilized for this purpose and will be discussed under Particulate Removal.

Solid adsorbents such as activated carbon or silica gel are best applied to low concentrations of gases and have been successfully used on iodine and oxides of nitrogen. Since adsorptive capacity is approximately 25 to 30 per cent of initial weight of material, an estimate of equipment size and life can be readily obtained. The adsorptive capacity of charcoal is approximately twice that of silica gel, and the latter also shows a preference for water vapor, which charcoal does not. Regeneration by steam stripping or heating is technically feasible if the units can be taken off stream temporarily.

The recovery of rare gases on a continuous basis requires low-temperature adsorption. Activated charcoal is used and a low-temperature refrigerant such as liquid nitrogen is used to cool the adsorbent and gases to be treated. Because of the need for such extremely low temperatures, the high cost of handling large gas volumes is apparent. Any size of system beyond a few cubic meters of gas would require a sizable investment in liquid-nitrogen supply or air-fractionation columns. This method has therefore been used only for limited volumes. If other methods of gas concentration could be employed such as concentration and stripping from a solvent, costs could be reduced, but such methods are not yet available.

The problem of **gases created by incineration or combustion** of solid wastes does not produce large amounts of radioactive gases. Gases to be expected would result from combustion of solids, liquids, or gases tagged with C-14, S-35, or I-131. Dilution appears to be satisfactory after removal of particulates which will be discussed below. Where gases have been of concern, the use of ethanolamine scrubbing towers for C-14 O₂ or S-35 O₂ has been studied and gives performances comparable with those obtained with nonactive isotopes.

* Magill, P. L., F. R. Holden, and C. Ackley, "Air Pollution Handbook," McGraw-Hill, New York, 1956.

DISPERSION AND REMOVAL OF AEROSOLS

Atmospheric dispersion of particulates by **stacks** follows most of the principles discussed above in reference to gases. Some modifications are necessary, however, for the case of coarser materials. Baron, Gerhard, and Johnstone * developed modifications of Sutton's equations to take account of the finite settling velocity of aerosol particles. Below 5μ or with terminal velocities of 0.15 ft/min (0.07 cm/sec) the effects of particle size are negligible for practical purposes. Above this value it should be understood that a decrease in atmospheric turbulence increases the maximum rate of deposition and the maximum concentration of the cloud at ground level. A decrease in turbulence shifts the maximum deposition point and ground concentrations downward from the source. For coarser aerosols ($>5 \mu$) the **maximum deposition** varies inversely with stack height (whereas for gases it changes inversely as stack height squared).

The maximum deposition at ground level for dry conditions can be computed at a distance x downwind from the formula

$$\omega_{\text{dry max}} = \frac{nQ}{(2\pi)^{1/2} C_y x^{2-n/2}} \quad (22-8)$$

where ω is the deposition rate (grams per square meter per second) for a steady source (Q in grams per second) or total deposition (grams per square meter) for an instantaneous source (Q in grams). Deposition-rate formulas other than maximum from which Eq. (22-8) is derived are presented in the AEC Handbook. †

To ascertain washout maximum from rain or precipitation the equation is modified as

$$\omega_{\text{rain max}} = \frac{Q}{(e\pi)^{1/2} C_y x^{2-n/2}} \quad (22-9)$$

where all terms are as in Eq. (22-8). From these equations some estimate may be made as to the hazard of deposited material, particularly in estimating for long-lived gamma emitters the conditions of greatest hazard for locations at large distances from the source.

Total instantaneous washout from a continuous source may be calculated from

$$\omega = \frac{Q}{(2\pi)^{1/2} C_y \bar{u} x^{(2-n)/2}} \quad (22-10)$$

Here ω = ground deposition in grams or curies per square meter.

Removal of Particulates or Aerosols

For the **removal of particulates or aerosols** created by nuclear-energy operations, air- and gas-cleaning equipment of most commercially available types has been used. These range from simple **inertial collectors** to **high-efficiency filters** or **elec-**

* Baron, T., E. R. Gerhard, and H. F. Johnstone, Dissemination of Aerosol Particles Dispersed from Stacks, *Ind. Eng. Chem.*, vol. 41, p. 2403, 1949.

† "Meteorology and Atomic Energy," U.S. Atomic Energy Commission, GPO, July, 1955. (Also AEC Rept. AECU-3066.)

Table 22-5. Operational Characteristics of Air-cleaning Equipment

Type of equipment (1)	Particle size range, mass median, μ (2)	Efficiency for size in Col. (2), % (3)	Velocity, ft./min (4)	Pressure loss, in. of water (5)	Approx cost/ cu ft./min (6)	Current application in AEC program (7)
Simple settling chambers.....	>80	60-80	25-75	0.2-0.5	\$0.05	Rarely used
Cyclones, large diam.....	>5	40-85	2,000-3,500 (entry)	0.5-2.5	\$0.10-\$0.25	Precleaners in mining, ore handling, and machining operations
Cyclones, small diam.....	>5	40-85	2,500-3,500 (entry)	2-4.5	\$0.25-\$0.50	Same as above
Mechanical centrifugal collectors.....	>5	20-85	2,500-4,000		\$0.20-\$0.35	Incorporated in chip traps for metal turning
Baffle chambers.....	>5	10-40	1,000-1,500	0.5-1.0	\$0.05	Rarely used, occasionally as cooling for hot gases
Spray washers.....	>5	20-40	200-500	0.1-0.2	\$0.10-\$0.20	Used on laboratory hoods and chemical separation operations
Wet filters.....	Gases and mists	90-99	100	1-5	\$0.05-\$0.10	Gas absorption and precleaning for acid mists
Packed towers.....	0.1-20 μ	90	200-500	1-10	\$0.40-\$0.80	Gas absorption and precleaning for acid mists
Cyclone scrubber.....	Gases and soluble particles <5	40-85	2,000-3,500 (entry)	1-6	\$0.25-\$0.40	Pyrophoric materials in machining and casting operations; mining, and ore handling; roughing for incinerators
Inertial scrubbers, power-driven.....	8-10	90-95		3 to 5 hp per 1,000 cu ft./min	\$0.15-\$0.25	Pyrophoric materials in machining and casting operations; mining, and ore handling; roughing for incinerators
Venturi scrubber.....	>1	99 for H ₂ SO ₄ mist; SiO ₂ oil smokes, etc., 60-70	12,000-24,000 at throat	0.03-0.15	\$0.50-\$3.00	Pyrophoric materials in machining and casting operations; mining, and ore handling; incorporated in air-cleaning train of incinerators
Viscous air-conditioning filters.....	10-25	75-85	300-500	0.1-0.3	\$0.004-\$0.006	General ventilation air; precleaning from chemical and metallurgical hoods
Dry spun-glass filters.....	5	85-90	30-35		\$0.02-\$0.04	General ventilation air; precleaning from chemical and metallurgical hoods
Packed beds of graded glass fibers, 1-20 μ , 40 in. deep.....	<1	99.90-99.99	20	10-30	\$1.0-\$5.0	Dissolver off-gas cleaning
High-efficiency cellulose-asbestos filters.....	<1	99.95-99.98	5 through media; 250 at face	1.0-2.0	\$0.04-\$0.06	Final cleaning for hoods, glove boxes, reactor air, and incinerators
All-glass web filters.....	<1	99.95-99.99	5 through media; 250 at face	1.0-2.0	\$0.07-\$0.10	Same as above
Conventional fabric filters.....	>1	90-99.9	3-5	5-7	\$0.30-\$1.00	Dust and fumes in feed materials production
Reverse jet fabric filters.....	>1	90-99.9	15-50	2-5	\$0.30-\$1.00	Dust and fumes in feed materials production
Single-stage electrostatic precipitator.....	<1	90-99, 90-95 on metallurgical fumes	200-400	0.25-0.75	\$0.50-\$2.00	Final cleanup for chemical and metallurgical hoods; uranium machining
Two-stage electrostatic precipitator.....	<1-5	85-99	200-400	0.25-0.50	\$0.25-\$0.50	Not widely used for decontamination

First, M. W., et al., Air Cleaning Studies, AEC Progr. Rept. NYO-1581, June 30, 1951.

First, M. W., et al., Air Cleaning Studies, AEC Progr. Rept. NYO-1580, June 30, 1952.

Friedlander, S. I., Silverman, P., Drinker, and M. W. First, Handbook on Air Cleaning, U.S. Atomic Energy Commission, September, 1952.

Blawie, A. G., and B. F. Judson, Filtration of Radioactive Aerosols by Glass Fibers, Air Rept. vol. 4, p. 223, 1955.

Silverman, L., Ched in Kershaw, M. G., Panel Discussion, Laboratory Design for Handling Radioactive Materials, BRAB Conf. Rept. 3, Building Research Advisory Board, National Research Council Washington, D.C., 1951.

Dennis, R., G. A. Johnson, M. W. First, and L. Silverman, Performance of Commercial Dust Collectors (Report of Field Tests), AEC Rept. NYO-1588, November, 1953.

First, M. W., et al., Air Cleaning Studies, AEC Progr. Rept. NYO-1581, June 30, 1951.

Dennis, R., et al., Dust Collectors Tested in Field, Chem. Eng., vol. 39, p. 196, February, 1952.

Dennis, R., et al., Dust Collectors Tested in Field, Chem. Eng., vol. 61, p. 187, May, 1954.

Johnson, G. A., et al., Performance Characteristics of Centrifugal Scrubbers, Chem. Eng. Progr., vol. 51, p. 176, April, 1955.

Scharnau, C. J., Design and Performance of Cyclone Dust Separators, Trans. Inst. Chem. Engrs., vol. 29, p. 356, 1951.

Scharnau, C. J., Design and Performance of Modern Gas-cleaning Equipment, J. Inst. Fuel, vol. 8, February, 1956.

trostatic precipitators. Modifications and technical variations of these devices have also been made as needed for special cases. A summary of the usual applications and characteristics of many of the devices employed is presented in Table 22-5. Some of the methods given are used in combination or series, and in some instances they are similar to devices widely used for other industries. Exceptions are the higher-efficiency final filters and prefilters and deep beds of fibers. The **high-efficiency filter** * was developed for special applications as required by the U.S. Army Chemical Corps and U.S. Atomic Energy Commission and is discussed elsewhere. Specific and extended data on performance of the various devices are given in the references as a footnote to Table 22-5.

A distinguishing feature of atomic-energy applications in most operations (except refining) is the low mass concentrations of materials to be collected. As described previously, this is apparent when the quantities involved are understood. In certain instances this simplifies control, and in other cases it can create complications. Loadings usually may be in the range of 1 to 10 times that of outdoor air (0.02 to 2 grains/1,000 cu ft of air or 0.05 to 5 mg/cu m) except for refining-process effluents. Equipment life for certain applications is therefore manyfold as compared with usual process wastes. Certain cleaning approaches are therefore feasible which would be short-lived in industrial-process loadings; that is, cleaning *in situ* is not always necessary. In reactor or reactor-shield air or gas cooling, it is essential to exclude particulates from the inlet air to prevent deposition as well as induced activity although most of the components of atmospheric dust, when irradiated, have short half-lives. Proper cleaning maintains the initial heat-transfer characteristics of the surfaces. It is desirable to clean as close to the source of contamination as possible to minimize deposition and accumulation in dust and piping. In some instances these become radiating sources requiring shielding. Cleaning close to the source may involve problems of high temperature and in the case of corrosive gases requires construction of special materials.

TYPES OF EQUIPMENT FOR PARTICULATE REMOVAL

Table 22-5 summarizes the various types of equipment for general air cleaning and their usual applications; only a few further comments are necessary for the inertial, wet, and electrostatic types. None of these devices except special precipitators operating in series at high voltages and with long retention times will give the 10^4 to 10^6 decontamination factors necessary in many instances for active materials. Their major disadvantage is that they cannot "fail safe"; that is, if power fails or air flow is impeded, either their efficiency is lowered or completely impaired performance results. Filtration devices, on the other hand, will perform well at values below design flow and become of serious concern only if holes or leaks develop within the media or if their resistance approaches values which seriously reduce air or gas flows.

Some comments in regard to dry and wet collection in general are necessary. The use of either implies some additional disposal problems. In high-efficiency final collection, dry procedures are always used. A summary of the advantages and dis-

* Smith, W. J., and E. Stafford, Development of a Dry Fibrous Filter, in McCabe, L. C., "Air Pollution," p. 64, McGraw-Hill, New York, 1952.

Table 22-6. Dry vs. Wet Collection

<i>Dry</i>	<i>Wet</i>
<u>Advantages:</u>	<u>Advantages:</u>
Recovery of material as dry product without alteration or further treatment Freedom from corrosion problems in most instances	Can collect gases and particulates simultaneously Recovers soluble materials readily. Material may be pumped to location for reuse concentration or disposal
Less storage capacity necessary for waste. Combustible filters may be employed for radioactive wastes to minimize storage	High-temperature gases quickly cooled and washed
Insoluble materials greater than 0.05 μ may be collected with high efficiency and long life of equipment	Corrosive gases and mists recovered
<u>Disadvantages:</u>	<u>Disadvantages:</u>
Certain products or hygroscopic materials may "cake" and cannot be easily removed	Soluble particulates for recovery must be recrystallized and may become contaminated by other materials during collection
Often requires secondary ventilation control to handle product or filter (viz., exhaust from bagging machine needs ventilation hood reintroducing dust to control system)	Requires wet disposal, sludge ponds, tailing piles, etc. These create a source of secondary contamination
Replacement or removal of filters may create an exposure problem to operators	Insoluble-product recovery requires wet-filtration equipment
High-temperature recovery requires special refractory materials or pretreatment of filter fibers	Particles less than 1 μ not readily wetted in most instances
Cannot be used in most cases for acid or corrosive mists	Corrosion problems
Can handle only gases or vapors which can be readily adsorbed or absorbed on a solid (e.g., activated charcoal, silica gel, soda lime, or soda ash)	Liquid entrainment in effluent air stream carrying contamination
<u>Common to Both Types of Collectors:</u>	Freezing problems in cold weather
Types with low and constant resistance Units with efficiencies approaching 100% Devices with automatic cleaning Equipment which will operate wet or dry Neither kind can remove rare gases without low-temperature refrigerants such as liquid nitrogen	

advantages of each type is presented in Table 22-6. The treatment of liquid radioactive wastes is an involved problem, especially if soluble salts or compounds are present which cannot be readily or economically precipitated. This subject is treated elsewhere in this handbook (Sec. 21). Liquid wastes or sludges from wet collectors can be piped to treatment areas and may require shielding in some instances. Criticality may also become a problem in wet collectors which recirculate significant collected materials in suspension.

DRY FILTERS

The first high-efficiency **asbestos-cellulose filter** was evolved from gas-mask and protective-shelter development. An improved form with increased air-flow capacity and life was developed for nuclear applications. This filter is made in several sizes for operations ranging from individual glove boxes to the 1,000 cu ft/min unit. Sizes commercially available are given in Table 22-7. There are four United States manufacturers and three foreign producers at the present time.

Table 22-7. Available Commercial Sizes of High-efficiency Pleated Filters

Designation	Dimensions, in.	Rated flow		Resistance at rated flow, in. H ₂ O
		cu ft/min	cu m/min	
A	8 by 8 by $3\frac{1}{16}$	35	1	1
B	8 by 8 by $5\frac{7}{8}$	50	1.4	1
C	24 by 24 by $5\frac{7}{8}$	500	14.1	1
D	24 by 24 by $11\frac{1}{2}$	1,000	28.3	1

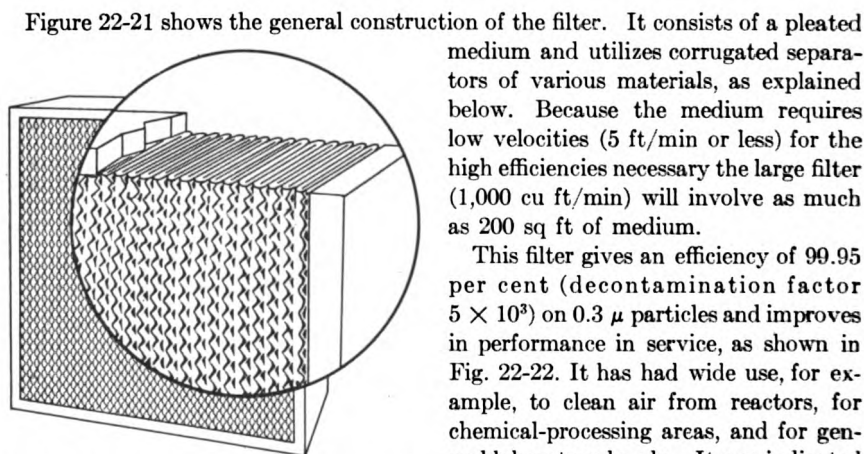


FIG. 22-21. Cellulose-asbestos filter as used for high-efficiency filtration of air. Unit shown is 24 by 24 by 6 in. and filters 500 cu ft of air per minute.

Figure 22-21 shows the general construction of the filter. It consists of a pleated medium and utilizes corrugated separators of various materials, as explained below. Because the medium requires low velocities (5 ft/min or less) for the high efficiencies necessary the large filter (1,000 cu ft/min) will involve as much as 200 sq ft of medium.

This filter gives an efficiency of 99.95 per cent (decontamination factor 5×10^3) on 0.3μ particles and improves in performance in service, as shown in Fig. 22-22. It has had wide use, for example, to clean air from reactors, for chemical-processing areas, and for general laboratory hoods. It was indicated in early use that the filter should not be used beyond resistances exceeding 2 in. (50 mm) of water because of excessive

power consumption and possible structural failure by rupture of the filter element. Subsequent work has shown that in practice filters can be used at resistances exceeding 8 to 10 in. of water. This added resistance will provide increased life. In many instances the added power cost is more than offset by the high cost of replacement of active units, and it is more economical to select design fan conditions to permit the higher resistances to develop in service.

It is advantageous to protect the filter and increase its life with a **prefilter**. With a spun-glass prefilter a significant improvement in life has been obtained. The

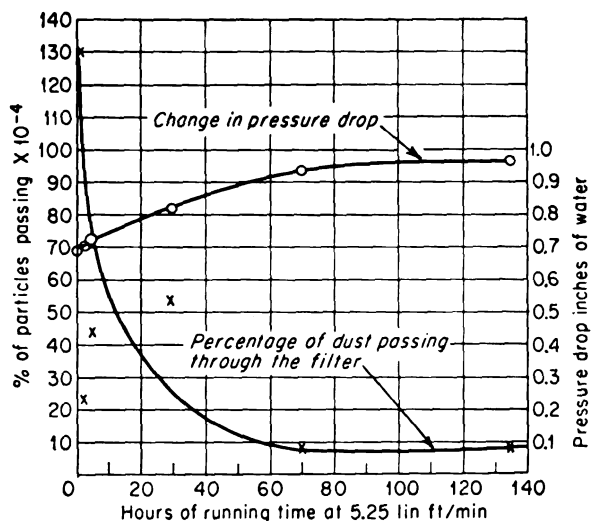


FIG. 22-22. Performance data for high-efficiency filter over prolonged period on atmospheric dust.

operating life of these filters in service has proved to be much greater than originally expected. Many have been in service for 3 years or more. In most instances changing has been necessary on a resistance basis rather than from activity on the filter.

The cellulose-asbestos filter is unable to withstand more than small amounts of corrosive gases and is limited to temperatures below 100°C. The medium is not fireproof, although it can be made fire-retardant. Danger is inherent with ignition and subsequent dispersal of retained contaminants. Pyrophoric materials will cause damage at their point of contact. In reactor-gas cleanup, spray cooling is used to reduce gas temperatures, and prefilters precede the final units. Very clean water is necessary to prevent the spray from forming an aerosol which will shorten filter life.

Because of the obvious limitations of cellulose-asbestos filters, an **all-mineral filter** was developed. Several types of medium are available. One is a 10-mil (0.01 in. or 0.25 mm) all-glass medium composed of all less than 1 μ (approximately 0.5 μ) glass fibers (MSA 1106 B) and another combination of 1.5 μ glass fibers and asbestos. These media have been fabricated into units similar to Fig. 22-15 but utilizing noncombustible separators of aluminium foil or mineral fiber and metal,

ceramic, or composite frames. Some have also been made without separators. The mineral filters give comparable or better performance than the paper media, as reported by Decker and his associates.*

With glass media and plastic cements these filters will resist most mineral acids. Exceptions are hydrofluoric, phosphoric, and alkali. Plastic fibrous media have been developed which can be used in such exposures.

For extremely high temperatures (1000°C) a ceramic fiber of 50 per cent Al_2O_3 and 50 per cent SiO_2 (Fibrafrax) has been made into webs for high-efficiency cleaning.

Glass fibers are also used in extended-life graded-layer filters for handling dissolver off gas or vessel vent gases. This application incorporates chemically resistant glass, in various diameters from 30 to 1 μ placed in layers of varying density and thickness. Table 22-8 shows a typical formulation as reported by Blasewitz and Judson.† Full-sized installations have been in service for over 3 years and give

Table 22-8. Filter Formulation for Cleaning Dissolver Off Gases *

(Superficial velocity approximately 20 ft/min. Gas stream contains varying amounts of submicron particles and acid gases up to 50% in volume)

Layer	Type of fiber glass	Fiber size, μ	Packing density, lb/cu ft	Filter-layer thickness, in.	Initial efficiency, %	Initial resistance, in. of water	Life expectancy ϵ , grains/sq ft †
Initial.....	115K	30	1.5	12	39	0.10	<1,400
Second.....	115K	30	3.0	10	53	0.24	<362
Third.....	115K	30	6.0	20	93	1.34	<228
Cleanup or final.....	AA	1.3	1.2	1	99.9	2.20	<28
Total.....		...	...	43	99.99	3.88	<1,400

* Blasewitz, A. G., and B. F. Judson, Filtration of Radioactive Aerosols by Glass Fibers, *J. Air Pollution Control Assoc.*, vol. 4, p. 223, 1955.

† ϵ is defined as the number of grains of methylene-blue aerosol ($M_g = 0.6\mu - \sigma_g = 2.4$) passing to the filter layer per square foot of filter area to produce a resistance increase of 4 in. of water.

low resistance and long life on low loadings of radioactive aerosols.

Filters of sand and gravel in graded layers were used for cleaning general cell-ventilation exhaust air before fiber-glass data were available. The greater porosity and small size of fibers with glass media reduce the physical size of the units for comparable performance. Since this type of filter unit becomes quite active it is placed in a concrete-lined pit on porous supporting-tile ductwork. An area large enough to handle the total gas volume at velocities of 20 ft/min (10 cm/sec) is used. Fifteen to twenty years of life at extremely high efficiency will be obtained before the filter is abandoned and buried in place.

Performance of fibrous media as used for final filters is given in Table 22-9. Here

* Decker, H. M., J. B. Harstad, F. J. Piper, and M. C. Wilson, Filtration of Microorganisms from Air by Glass Fiber Media, *Heating, Piping, Air Conditioning*, vol. 26, p. 155, 1954.

† Blasewitz, A. G., and B. F. Judson, Filtration of Radioactive Aerosols by Glass Fibers, *J. Air Pollution Control Assoc.*, vol. 4, p. 223, 1955.

Table 22-9. Performance and Ratings of High-efficiency Dry Filters

(At normal air temperatures and standard-density air)

Medium	Test aerosol		Air velocity		Resistance, in. of water	Efficiency, %	Method	Remarks	Ref.
	Name	Size, mg (homogeneous except *)	ft/min	cm/sec					
CC-6 cellulose-asbestos paper	Methylene blue.....	0	4	2	0.8	99.9871	Discoloration		1
	Diethyl phthalate (DOP).....	0.3	5	2.5	0.67	99.85	Penetrometer		2
	Atmospheric dust.....	0.5 *	5	2.5	0.67	99.9+	Count	Note excessive velocity causes greater penetration of fine size	3
AEC No. 1 cellulose-asbestos	Duraluminum.....	0.18	500	250	100	97.7	Count	Reduced velocity improves performance	3
	Duraluminum.....	0.18	2	1	0.28	99.7	Count	Size for max penetration	4
	Potassium permanganate, KMnO ₄	0.02	20	10	2.7	93.0	Penetrometer		2
All-glass superfine fibers, Hurlbut MSA 1106B	DOP.....	0.3	5	2.6	0.7	99.78	Count		3
	Duraluminum.....	0.18	40	20	5.6	99.6	Count		3
	Atmospheric dust.....	0.5 *	5	2.5	0.7	99.98	Count		2
Sand.....	KMnO ₄	0.01-0.02	4	2	0.56	91.0	Penetrometer	Size for max penetration	4
	DOP.....	0.3	5	2.5	1.05	99.999	Count		2
	Atmospheric dust.....	0.5 *	5	2.5	1.05	99.9+	Count	Size for max penetration	4
Composite glass wool.....	KMnO ₄	0.015	20	10	4.4	93.0	Radioactivity	Size for max penetration	4
	Cell-ventilation gases.....	1 μ	3-5	1.5-2.5	4.5-5.5	99.5-99.8	Count	69 in. deep, graded sizes from 2 1/4 to 50 mesh. Will not withstand high moisture conditions	5
	Process off gases.....	1 μ	20	10	4.0	99.9+	Radioactivity	Composition given in Table 22-8	5
Compressed glass fibers.....	Methylene blue.....	0.6 μ MMP	20	10	4.0	99.99	Gravimetric	Same as above	5
	Atmospheric dust.....	0.5 μ	5.25	2.6	0.69	99.987	Count	0.02 in. thick, 50% 1.3 μ and 50% 3.0 μ fibers	6
Resin wool.....	Atmospheric dust.....	0.5 μ	14	7	0.3	99.6	Discoloration	These filters are known to decrease in performance when exposed to ionizing radiation	7
	Uranium oxide.....	0.12 μ	2.3-7.8	1.2-3.9	0.30-1.23	95.5-99.5	Gravimetric	Special glass formulation developed by A. D. Little. Aluminum separators and furnace-cement seals	8

1. Tench, F. M., Operating Experience for Pile Filter House, Compilation of papers presented at the meeting of the Stack Gas Problem Working Group, June 21, 22, 1949, Washington, D.C., Technical Information Division, ORE, Oak Ridge, Tenn.
 2. Smith, W. J., and N. F. Suprenant, Properties of Various Filtering Media for Atmospheric Dust Sampling, *Proc. Am. Soc. Testing Materials*, vol. 53, p. 1122, July, 1953.
 3. Fitzgerald, J. J., and C. G. Detwiler, Collection Efficiency of Air Cleaning and Air Sampling Filter Media, *Am. Ind. Hyg. Assoc. Quart.*, vol. 16, p. 122, 1955.
 4. Fitzgerald, J. J., and C. G. Detwiler, Collection Efficiency of Filter Media in the Particle Size Range of 0.005 to 0.1 Micron, *Am. Ind. Hyg. Assoc. Quart.*, vol. 18, p. 47, 1957.
 5. Blasewitz, A. G., and B. I. Judson, Filtration of Radioactive Aerosols by Glass Fibers, *J. Air Pollution Control Assoc.*, vol. 4, p. 223, 1955.
 6. A. D. Little, Inc., Investigation of Stack Gas Filtering Requirements and Development of Suitable Filters, AII-23, Rept. 8 to Division of Engineering, AEC, Feb. 28, 1950.
 7. Billington, N. W., and D. W. Saunders, Air Filtration, *J. Inst. Heating Ventilating Engrs.*, vol. 15, p. 46, 1947.
 8. Connors, E. W., Jr., and D. P. O'Neil, Efficiency Studies of a High Efficiency-High Temperature Filter against Freshly Generated Uranium Oxide Fume, *Argonne Natl. Lab. Rept. ANL-3453*, June, 1954.
- Additional references on high-efficiency filters:
 Stafford, E., and W. J. Smith, Dry Fibrous Air Filter Media, Performance Characteristics, *Ind. Eng. Chem.*, vol. 43, p. 1346, 1951.
 Silverman, L., Performance of Industrial Aerosol Filters, *Chem. Eng. Progr.*, vol. 47, p. 462, 1951.

sand and AEC types are also given for comparative purposes. Newer data on sub-micron aerosols indicate their limitations for very small particles.

Filters used where resistance increases rapidly in service and which become highly radioactive must be handled away from personnel. A variable-compression filter shown in Fig. 22-23 has been developed for this purpose.* Filter resistance

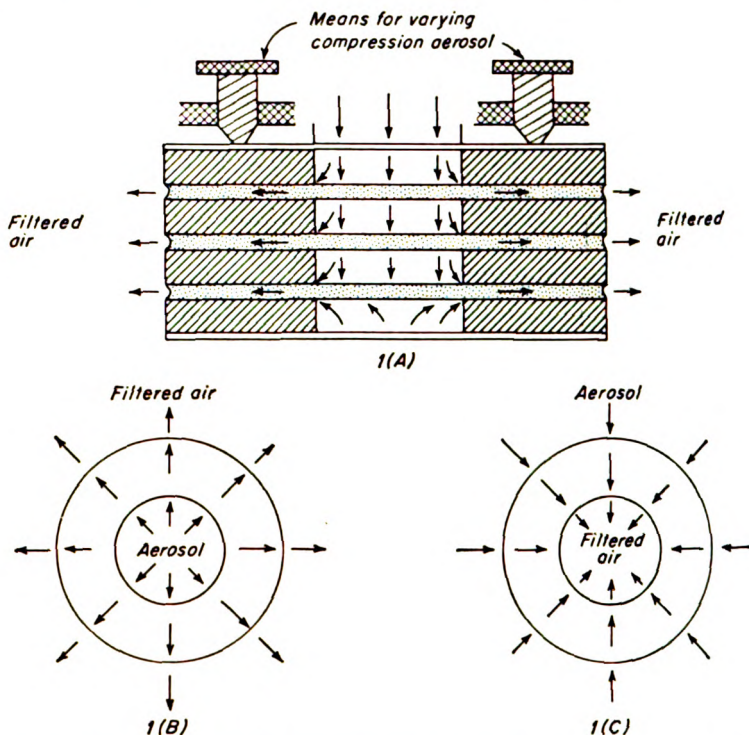


FIG. 22-23. Principle of variable-compression filter as developed for remote control of resistance and utilizing a bonded fiber-glass medium.

can be controlled remotely by means of a mechanical linkage or a motor drive. The unit incorporates compressed layers of resin-bonded spun fiber glass (1.5 or $3\ \mu$) arranged so they can be gradually released in compression as resistance increases. Collected particulates thus penetrate farther into the media. Efficiencies well above 95 per cent can be obtained at reasonable resistances. The phenolic-bonded flexible fiber glass gives resilience to maintain the shape of the medium.

It has been a useful device in small sizes (< 100 cu ft/min), comparable with the pilot unit investigated by the authors as an inlet filter for glove boxes. The unit can readily be adjusted to control the negative pressure within the box.

One important application for refining operations may be mentioned, and that is

* Silverman, L., and M. W. First, Edge and Variable Compression Filters for Aerosols, *Ind. Eng. Chem.*, vol. 44, p. 2777, 1952.

the use of continuously or automatically cleaned pressed-wool or synthetic felts.* These are made in the form of bag units which are cleaned by high-velocity jets. Rotary-arm and disk-type units are two new forms which appear useful for heavy-loadings applications. The major advantage of this type for refining fabrication operations where heavy loadings occur with radioactive metals is that constant resistance maintains uniform ventilation rates. The greater capacity in cubic feet per minute per square foot of cloth area (cu ft/min/sq ft or filter velocity) reduces the number of bags or compartments and minimizes maintenance and replacement. The principle of this unit is shown schematically in Fig. 22-24.

WET FILTERS

Mixed particulates as of acid mists and solids are common, and the fiber-glass devices mentioned above are applicable. The use of continuously wetted fiber beds and dry filters is another approach to solving this problem. Wet collection cannot be made so inherently efficient, because entrained moisture or misting creates effluents not readily captured. In general, wet collectors should therefore be followed by some dry-collection procedure.

Considerable study has been given to wet-cell units followed by dry pads.† This type of device has been used for cleaning of laboratory-hood effluents when connected to a central exhaust system. Studies show that this type of unit is an excellent gas-absorption system because of the extended wetted surface provided by the fiber-glass packs. A composite unit employing Dynel and Saran plastic fibers was developed for combined hydrogen fluoride gas and mist absorption.‡ This unit removes 99 per cent of hydrogen fluoride gas and 85 to 95 per cent of particulates in the $1\ \mu$ range. If fine radioactive particulates ($<1\ \mu$) are present, units should include a plastic-fiber-filter type comparable with cellulose-asbestos. Mixed particulates are also associated with radiochemical-laboratory use of perchloric acid. A small composite unit comprising a wetted cell, a mist eliminator, and an all-mineral filter has been developed for this purpose, as shown in Fig. 22-25. This device fits into typical laboratory hoods, as described.§ It is thus possible to eliminate all

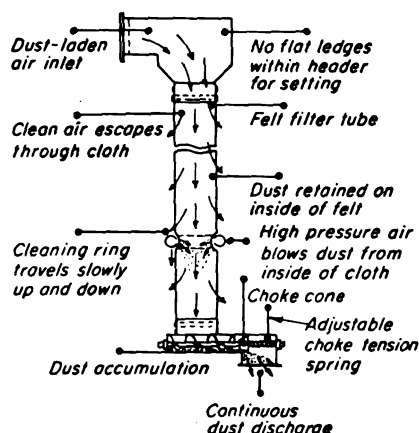


FIG. 22-24. Schematic diagram of reverse-jet filter unit.

* Friedlander, S., L. Silverman, P. Drinker, and M. W. First, "Handbook on Air Cleaning," U.S. Atomic Energy Commission, GPO, September, 1952.

† First, M. W., R. Moschella, L. Silverman, and E. M. Berly, Performance of Wet-cell Washers for Aerosols, *Ind. Eng. Chem.*, vol. 43, p. 1363, 1951.

‡ Berly, E. M., M. W. First, and L. Silverman, Recovery of Soluble Gas and Aerosols from Air Streams, *Ind. Eng. Chem.*, vol. 46, p. 1769, 1954.

§ Silverman, L., and M. W. First, Hood Scrubber Unit for Perchloric Acid, Harvard School of Public Health, *AEC Rept.* NYO-1589, Nov. 30, 1954.

organic materials in the cleaner with which perchloric acid might react. Removal eliminates the danger that perchlorates may form in ducts or on succeeding dry

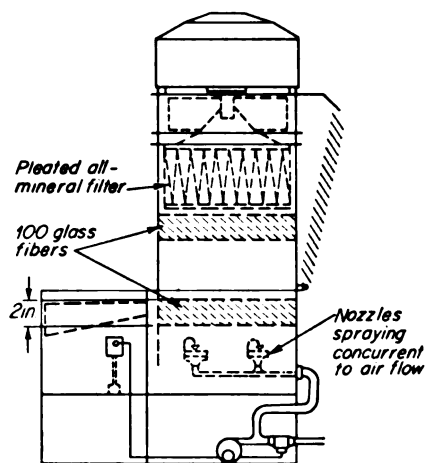


FIG. 22-25. Schematic diagram of perchloric acid hood-scrubber unit—combining spray-section wetted filter and dry final high-efficiency filter.

hood filters. With this unit, perchloric acid may be used in hoods provided with usual filters. Similar problems have arisen in interhalogen chemical use in uranium metallurgy. Since these reactions take place in protected hoods, additional treatment was necessary to remove the interhalogen decomposition products, such as hydrogen fluoride, chlorine, chlorine monoxide, chlorine dioxide, hydrogen bromide, and hydrogen chloride. Aluminum wire filters or crushed limestone packs can be used in hoods, but for larger quantities of materials a scrubbing system is necessary, as described above. In the perchloric acid scrubber described, the final all-mineral filter can be washed and cleaned several times, which removes the entrained acid. The washing process

does not remove particulates embedded in the fibers. It is apparent that fiber-glass media can filter saturated gas streams without difficulty.

INCINERATION AND EVAPORATION

Reducing solid and liquid wastes is of major concern. Reduction of the volume of low-activity waste to one-tenth by **incineration** or liquid waste to one-hundredth or to its solid content by **evaporation** has been investigated at many facilities. Incinerators must incorporate an optimum gas-cleaning system. High temperatures are developed (500 to 1000°C); therefore, it is necessary to provide cooling before particulate removal in most instances. Reducing gas volume by cooling or wet scrubbing minimizes the final cleaning equipment. Gas-cleaning equipment applied to incinerators may be spray towers with elaborate scrubbers, such as the venturi scrubber in series, and finally filtration through a spun-glass prefilter with a final filter of cellulose-asbestos. Reheating may be necessary to prevent condensation on filters. A serious difficulty associated with incinerator operation is that improper combustion results in tar formation. Carbonaceous particulates which are gummy or sticky may rapidly plug high-efficiency cleaners. Proper incineration will reduce some of the problems in the gas cleaning. A successful design has been reported by Rodger and Hampson.*

The problems created by evaporation of liquids as to aerosol formation depend

* Rodger, W. A., and D. C. Hampson, Operating Characteristics and Economics of a 100 Ft³/Day Incinerator for Radioactive Wastes, *J. Air Pollution Control Assoc.*, vol. 6, p. 41, 1956.

on the methods employed for reduction of liquid volumes. The highest efficiency in reducing volume is obtained by vapor-compression evaporation. Filtration of entrained mist is done by means of fiber glass. Manowitz and his coworkers * have studied vapor-compression equipment incorporating fiber-glass filters for reducing radioactive particulates. They were able to obtain 100-fold reduction in liquid-waste volume. Fiber-glass-layer composition is similar to those employed for air- or gas-filtration purposes for coarser aerosols. In evaporation of liquid wastes rapid boiling, or ebullition, should be avoided, as it produces droplets which may contain entrained solids; upon evaporation these particles become aerosols.

PERFORMANCE AND CHOICE OF EQUIPMENT

Of major concern in evaluation of air cleaners has been the method of sampling and rating performance. To obtain an over-all evaluation requires more than a single test aerosol. A single aerosol is useful for rapid rating of high-efficiency filters where each must be tested to meet specific requirements. The DOP (dioctyl-phthalate) condensation aerosol of $0.3\ \mu$ is nearly homogeneous and offers a simple test substance. Rapid measurement of concentration is done with a light-scattering (forward) photometer. Since DOP is a liquid droplet, however, results are not the same as obtained with solid particulates. For potential nuclear-energy applications of devices or methods, the author uses a number of aerosols made by dispersing of solid particulates or by spraying liquids containing dissolved salts (such as copper sulfate or uranium nitrate) which can be made into solid particulates of controlled size. Liquid smokes from oil-vapor condensation or burning tobacco are also employed. The size of the aerosol series used ranges from less than $0.1\ \mu$ (vaporized metals) to particles in the 1 to $5\ \mu$ range made from redispersed reduced solids, such as talc, fly ash, calcium carbonate, and other mineral dusts. For dust testing with high-specific-gravity materials, iron-powder spheres prepared from iron carbonyl may be used. Performance should be based on gross efficiency by weight, or efficiency for a specific particle size may also be utilized. Stain or discoloration evaluation, and for extremely toxic aerosols the number of particles penetrating is selected. A complete description of test methods and aerosols is given by the author in another handbook.†

Methods of sampling for radioactive aerosols from which collector performance can be determined are discussed elsewhere in this handbook.

CLEANING GASES FROM POWER REACTORS

Particulates from power reactors come from two major sources. These are identified with conditions at the inlet and the outlet of the reactor-cooling air or its shield-cooling air. The cleaning of air entering the reactors minimizes deposition and induced radioactivity. The degree of cleaning necessary for gases leaving the reactor depends upon the reactor design and the amount of precleaning. Regardless

* Manowitz, B., P. Richards, and H. Horrigan, Vapor Compression Evaporation Handles Radioactive Waste Disposal, *Chem. Eng.*, vol. 62, p. 194, 1955.

† Magill, P. L., F. R. Holden, and C. Ackley, "Air Pollution Handbook," McGraw-Hill, New York, 1956.

of the reactor design, there is likely to be gas generation somewhere in the system, with possible entrainment of particulates. In present reactor designs some portion of the system may involve gas or air contamination. Certain package types may not involve gas in quantities until reprocessing of fuel elements. In cleaning reactor-produced gases and particulates, the methods which have been described above can be applied, and the importance of reducing the gas volumes involved cannot be overemphasized.

The problems of cleaning associated with **air-cooled reactors** are more acute if rupture of fuel elements takes place in operation or if serious erosion of surfaces results. The important feature is that loadings will be low, unless a large number of fuel elements fail at once. Reactor systems must be safeguarded against large numbers of element ruptures. Should a catastrophic or multiple-failure reaction take place, the downstream filter should be capable of retaining most of the volatilized material. Devices which will permit cleaning of gases at high temperature are necessary. Research is now directed toward developing devices which will operate continuously at temperatures above 500°C. Stacks are more effective at higher temperatures for dispersion of radioactive gases and any discharged particulates.

Special problems may arise in decontaminating radioactive aerosols which are unique to the nuclear industry. These may be listed: (1) variations in processes to which installed equipment may be exposed; (2) decay products of noble gases; (3) handling of collected aerosols or contaminated filters; (4) radioactive liquid or solid waste disposal; (5) monitoring of effluents; and (6) maintenance of cleaning equipment.

The problems with **process variations** in this instance are associated with the changing from one type of operation conducted in a given hood or glove box to another in which a different type or size of aerosols is created. The cleaning system should be reevaluated for such cases to determine if it will be suitable. For example, the use of cellulose-asbestos filter on uranium or plutonium extractive procedures employing interhalogens may cause serious problems. These gases are directly reactive or can break down into highly reactive gases or vapors. A similar situation can arise from pyrophoric materials which can ignite or burn holes in media. Mixed aerosols involving chemically reactive gases or solvents should also be given attention before their generation in a system incorporating a cleaner designed and selected for another contaminant.

DECAY PRODUCTS OF NOBLE GASES

Rare-gas isotopes such as Kr-89, Kr-91, Xe-141, and all others, it is believed, with a half-life of more than 3 sec decay to radioactive particulates. In the original air-cooled Oak Ridge reactor, for example, particulate decay for 100,000 cu ft/min of effluent cooling air is estimated as about 1 mc/day of long-lived beta emitters or 5 to 10 mc/day of short-lived activity. These particles are less than 0.2 μ when formed. If in sufficient number they may agglomerate or coalesce with other aerosols present. Because of their formation after passing through a filter, the actual efficiency of a fiber-glass, cellulose-asbestos, or high-efficiency filter combination is less than 99.95 per cent. Actual measurement indicated values of 91 per cent removal of gross beta activity. Because of the time delay and particle size involved

it is obvious that atmospheric dilution with a stack is the only reasonably economic solution at present with large gas volumes containing such products.

ASSOCIATED PROBLEMS

Because of their severe cleaning requirements, highly efficient filters must be properly sealed and gasketed to obtain the decontamination values cited. When **contaminated filters** are to be removed it is usually desirable, especially with highly toxic alpha emitters, to fix the activity on the filter surface. This is usually done while they are still installed by spray coating them with an acrylic spray or "cocoon" material before removal. For plutonium, auxiliary bagging devices have been developed which permit the filter to be lowered into a plastic bag without dispersing dust to the ambient air. The bag is heat-sealed, and the filter is placed in a paper carton, usually the original shipping container. They may be baled and stored, buried, or disposed of in an incinerator. Filters contaminated with active gamma material may have radiation levels that prevent easy access. A remote-control changing device has been developed * for this purpose which permits a hydraulic ram to force a clean unit into place and drop the active unit into a sealed chamber. For continued operations a bypass air-flow filter system is necessary.

The use of scrubbers or washers may involve the recirculation of **contaminated water or liquors**. It is important that pump seals and vessels be leaktight to prevent spillage and secondary aerosol formation. Liquid wastes can be piped to storage or decay vessels, or containers for treatment, evaporation, or disposal as described elsewhere in this handbook (see also Sec. 6).

The high degree of decontamination necessary for nuclear processes requires **stack-effluent monitoring** in order to maintain constant evaluation of performance. The samplers for this purpose are described elsewhere in this handbook. Continuous-recording devices for alpha, beta, and gamma activity as gases or particulates are available. Where large particles ($>5 \mu$) are present, isokinetic sampling probes are necessary. †

In the case of radioactive decontamination one cannot too strongly emphasize the need for continued and careful **maintenance of cleaning equipment**. Filter performance is highly dependent upon the preservation of tight seals and mineral leakage. Continued operation of exhausters and blowers can cause leakage to develop by vibration; hence monthly monitoring or even weekly checking is desirable. It is also essential to observe resistance increase at filters or cleaning devices, as excessive resistance will decrease air flows below satisfactory levels at the hood face or opening.

* Blomgren, R. A., and N. J. G. Bohlin, A Method of Changing Alpha and Gamma Contaminated Filters without Interrupting Exhaust, *AEC Rept. TID-7513*, pt. 1, p. 69, June, 1956.

† Dennis, R., W. R. Samples, D. M. Anderson, and L. Silverman, Isokinetic Sampling Probes, *Ind. Eng. Chem.*, vol. 49, p. 294, 1957.

Section 23

PERSONNEL CONTROL

By

JAMES H. STERNER, M.D., *Medical Director, Eastman Kodak Company,
Rochester, New York.*

WILLIAM L. SUTTON, M.D., *Laboratory of Industrial Medicine, Kodak Park
Works, Eastman Kodak Company, Rochester, New York.*

Selection of Personnel.....	23-2
Medical Control of Exposed Workers.....	23-6
Medical Records.....	23-12
Medicolegal Considerations.....	23-14

PERSONNEL CONTROL

James H. Sterner and William L. Sutton

SELECTION OF PERSONNEL

The **selection of personnel** for work involving exposure to ionizing radiation requires, in addition to the basic elements of occupational placement, a consideration of the special problems peculiar to radiation exposure in medical practice, radiation research, and industry. The nature of the radiation hazards, the fact that dangerous levels of exposure cannot be detected by the senses, the complex and difficultly comprehended character of radiation injury, and the confused and often misleading information which has been widely disseminated distinguish the radiation hazard from other occupational hazards.

The majority of tasks involving radiation exposure, particularly in the expanding peacetime application of nuclear energy as well as in the development of nuclear weapons, will have such a degree of technical control of radiation hazards as to require only minimal to moderate effort and understanding by the worker to assure adequate control. For this work, the selection of employees will differ little from that for industry in general.

For a limited number of jobs inherently involving a greater risk of radiation exposure, a higher degree of selectivity will be required. In some instances the hazard may be limited to the individual only, or to a small group of associates. In other cases, as, for example, with **operators of nuclear reactors** a serious mistake may involve many people and expensive plant and equipment not only in the installation but in the surrounding community. An adequate evaluation of the mental and emotional qualifications for these workers is of paramount importance.

The objectives of the **medical examination** for the selection of workers who may be subject to radiation exposure are the same as those for general industrial medical purposes: *

1. To measure the medical fitness of individuals to perform their duties without hazard to themselves or others.
2. To assist individuals in the maintenance or improvement of their health.
3. To detect the effects of harmful working conditions and advise corrective measures.
4. To establish a record of the condition of the individual at the time of each examination.

* Guiding Principles of Medical Examinations in Industry, *J. Am. Med. Assoc.*, vol. 161, pp. 975-978, July 7, 1956.

The significant specific factors which must be considered in the selection of personnel for work which carries a high risk of radiation exposure are:

1. The amount of previous radiation exposure.
2. Family history, especially as it pertains to hereditary defects.
3. Physical conditions similar to or related to the pathological effects of radiation.
4. The age of the worker.

The amount of **previous radiation exposure** may be difficult to establish, but the best possible estimate of the total exposure should be obtained. Even though the applicant had received a number of times the maximum permissible accumulated dose, the physical examination is unlikely to reveal any indicative findings.

Thus the work history, if it includes previous significant exposure to radiation, must be developed so as to obtain at least a quantitative guess at the accumulated dose. If there have been repeated **diagnostic X-ray exposures**, or **radiation therapy** for any cause, such estimated dosage should be carefully weighed in relation to the work exposure. If the accumulated dose cannot be reasonably determined, the assumption should be made that the applicant has received his maximum accumulated dose.

No arbitrary limits can be set as to acceptability of the applicant because of his accumulated dose. With the downward revision of the permissible limits, a limited number of individuals may be encountered who have exhausted their "reserve" exposure based on current criteria (see Sec. 3). Frequently these are the people who have developed the knowledge and skills which make them of greatest value in work involving radiation exposure. At present the hazard to the individual and to the employer must be balanced in each instance with the economic and technical need for a specific employment assignment. With the trend, in many states, toward the establishing of permissible occupational exposure limits (see Sec. 3), the regulation of employment in specific jobs may become mandatory.

The family history should be explored and recorded in more detail than is usually routine. Knowledge of the precise mechanisms of many conditions known or suspected to be hereditary is often fragmentary. Even less is known about more subtle defects such as decreased fitness and vitality. Nonetheless, a record of the occurrence of such inheritable defects or **congenital abnormalities** in an applicant or his family may be of some value should related abnormalities occur in his offspring. Particular attention should be directed toward recording neonatal deaths, visible malformations, mental defectives, microcephaly, and hematologic abnormalities in the family. A marital history is relevant—particularly information concerning both the applicant's and spouse's fertility, the wife's obstetrical experience, and the children's health. Except for high-risk occupations, where each case should be considered individually, the history of familial defects of this type would not usually be sufficient for rejection.

Certain physical conditions are indistinguishable from the **pathological effects of radiation**. In some instances, these conditions may be aggravated by radiation, with the consequent increased liability to the employee and employer. In others, it may be claimed, with progress of the disease condition, that radiation exposure was the cause. The nature and degree of such conditions must be evaluated individually in relation to the likelihood of further injury, and of valid or false compensation claims.

A not uncommon example is the finding of a thin atrophic skin with telangiectasis and areas of hyperkeratosis. This can result from excessive irradiation but is found in many cases where there was no radiation exposure. Since the cause in any one case cannot be differentiated, the likelihood of medicolegal complications is considerable. Other conditions which may become involving factors are:

1. Low normal or unstable hematologic findings (i.e., a low polymorphonuclear white-cell count).
2. Abnormalities of the lens of the eye, particularly beginning cataract formation.
3. Abnormal changes in the lungs, such as fibrosis revealed on X-ray examination—particularly where subsequent exposures may involve radioactive dusts or gases.
4. Diseases such as multiple sclerosis and muscular dystrophy, where the etiological factors are unknown or poorly understood.
5. Infertility and habitual spontaneous abortion.

The age of the applicant can be an important consideration, particularly for work which carries a higher risk of radiation exposure. With the limiting factor in establishing the permissible accumulated dose predicated upon genetic grounds, it is generally agreed that exposures incurred after age forty have less serious implications. Thus, where the risk of accidental overexposure is higher, consideration should be given to the use of older persons, particularly those whose "reserve" has not been depleted by earlier exposure to radiation.

The revised recommendations of the National Committee on Radiation Protection (NCRP) officially recognized that occupational radiation exposure is undesirable before the age of nineteen when it set the maximum permissible accumulated dose at 5 ($N - 18$) "where N is age and greater than 18." There have been no detailed official proposals concerning occupational exposure of women in the childbearing age. The NBS Handbook 59 states that "it is not necessary to recommend for pregnant women as such, a weekly dose lower than the basic permissible weekly dose." There is general agreement, however, that the embryo or fetus is highly vulnerable because of (1) the rapid growth; (2) the critical and delicate developmental changes taking place; (3) the small size, so that the proportion of the radiation to the mother received by the fetus will be "whole-body" radiation; and (4) the extreme youth—which affords a longer exposure period in the event of absorption of isotopes with long biological half-lives. This increased liability to injury is particularly significant in the earlier phases of fetal development when the mother is apt to be unaware of her pregnancy. In general, wisdom will dictate that women in this age group should not be exposed to high radiation risks.

Mental and Emotional Factors. For the majority of jobs involving exposure to radiation, operating conditions will be so regulated as to maintain radiation exposures well below permissible levels. With the increasing extension of regulatory controls by governmental agencies, the restrictions imposed on operations with higher potential exposures will stimulate and justify preventive measures to reduce radiation to a minimum. For these jobs, the usual standards of industrial employment will apply.

Certain operations will necessarily carry a higher potential risk, such as the assembly and testing of sources of radiation; short-term operations or technically complicated procedures where the technological or economic factors restrict the application of safeguards; many research projects involving new or unusual sources

or types of radiation; and the operation of nuclear reactors where, although the likelihood of accident is small, the consequences may be catastrophic. For these types of employment, special efforts must be made to select steady, responsible, emotionally mature, intelligent individuals.

The evaluation of the **mental and emotional capacities** should begin with the initial employment interviewer. In addition to the specific educational and training experience, much valuable information can be obtained from a careful history of the school and work experience which includes reasons for changes; record of absenteeism; reactions to work situation, associates, and supervisors; and social behavior including participation in groups, clubs, and other community activities.

The examining physician has the opportunity to develop this background information to include a history of childhood reactions, marital relations, and response to stress situations. A more formal assessment of personality traits may be obtained by the use of such tests as the Cornell Index (Mittelman, B., *et al.*, The Cornell Indices, The Cornell Word Form, *Ann. N.Y. Acad. Sci.*, vol. 46, p. 573, 1946), but these procedures have not been validated with respect to employment experience. The identification of characteristics such as excessive aggression, paranoid tendencies, wide swings of depressive or hypomanic behavior, lack of responsibility, and repeated involvement in accidents should signal caution.

Physical Requirements. In general, the **physical requirements** for employees with radiation exposure will be determined by the specific demands of the job other than the factor of radiation. The companies which have had considerable experience in the operation of plants involving radiation exposure apply employment standards for jobs in these plants which are similar to those used in other nonradiation industrial operations.

The emphasis which is placed on specific elements of the physical examination will depend upon the job demands. For example, the vision requirements of a panel-board operator for a nuclear reactor will be more stringent with respect to acuity, color vision, and peripheral vision than those needed by general maintenance workers. The necessity of taking action in response to an audible warning signal requires an adequate ability to hear. Manipulation of robot handling equipment demands a dexterity and coordination. The examining physician must be familiar with the job demands from personal knowledge or by having written descriptions of the job characteristics.

Special factors in the examination requiring consideration because of radiation exposure include:

1. The skin. Thin atrophic skin, with multiple telangiectases, hyperkeratoses, brittleness and ridging of the nails, diminished or loss of ridge markings of fingers, and any of the chronic dermatoses which are frequently treated by irradiation.
2. The blood. Hematologic values outside the normal range. There is no agreement as to absolute limits, but the following ranges may serve as a guide:

	Male	Female
Hemoglobin.....	13.5-18 g/100 ml	12-16 g/100 ml
Erythrocytes.....	4,500,000-6,500,000/cu mm	4,000,000-5,500,000/cu mm
Cell volume.....	40-54	37-47
Leukocytes.....	4,000-12,000/cu mm	
Neutrophiles.....	2,000-9,000/cu mm	
Eosinophiles.....	50-500/cu mm	
Monocytes.....	100-1,000/cu mm	
Lymphocytes.....	1,000-4,000/cu mm	

In addition, the finding of abnormal or excessive numbers of immature cells in the blood smear, a history of instability or wide fluctuation of hematologic elements or an apparent decrease or abnormal characteristics of platelets should be considered.

Each laboratory should develop sufficiently standardized techniques and analyze critically their results so that a normal or acceptable range can be established for their employment group. An unrealistic rejection level results in unjustifiable loss of normal people from the work force. In any cases questioned one or more repeat examinations should be made to establish a more accurate appraisal of the base-line hematologic values.

3. A history of radiation therapy. Such diverse conditions as enlarged thymus, eczema, osteomyelitis, Marie-Strumpel syndrome, and many others have been treated with intensive radiation therapy. An effort should be made to quantitate the amount of radiation received, and a painstaking examination should be made to evaluate and record any radiation sequelae. The likelihood of continuing or subsequent radiation therapy for the condition must be considered.

4. A history of exposure to radioisotopes. The conditions under which such exposures occurred, the controls applied, and the bio-assays performed should be determined. If there was a significant exposure, bio-assay determinations may be considered to determine the preemployment status.

MEDICAL CONTROL OF EXPOSED WORKERS

The **scope of the medical control program** should be related to the degree of exposure. There are no specific medical procedures which will indicate any effects from radiation exposure of the order of, or moderately exceeding, the maximum permissible dose from external sources of ionizing radiation, recommended by the NCRP. It is only when the amount of radiation has greatly exceeded the permissible dose levels that physical changes can be detected by presently accepted medical tests. Thus, while the value of the periodic examination is limited as a control procedure, it is an indispensable and final factor in evaluating the effect of the exposure.

The need for frequent **periodic medical examination** is in inverse proportion to the thoroughness, reliability, and effectiveness of the radiation monitoring and control program. Where the radiation levels encountered are at or below the permissible dose, and the likelihood of accidental overexposures is low, an annual

examination is recommended. The frequency with which "technical overexposure" (for example, doses larger than the pertinent maximum permissible dose for a given control period even though compensated by subsequent periods of no or minimal exposure) occurs determines the need for additional medical examination. When significant accidental overexposure has occurred, or is suspected, a complete examination is immediately indicated, with subsequent examinations to be performed depending on the character and degree of overexposure.

The specific type of medical examination which is indicated, both for routine periodic studies and for instances of accidental overexposure, is determined in part by the nature of the exposure—whether to external radiation or to internal emitters. With whole-body radiation from external sources, it is unlikely that any detectable effects will be demonstrated after an acute exposure of less than 25 r (NBS Handbook 59, p. 59, Sept. 24, 1954). The acute absorption of an internal emitter giving rise to an equivalent radiation dose similarly would be unlikely to be detected by routine medical procedures.

The periodic examination must be designed to fulfill the general objectives of **health examinations in industry** as well as to elicit specific findings of interest in persons exposed to ionizing radiation. The same principles that are involved in medical examinations for selection of personnel guide the conduct of the periodic examination. The serial character of the periodic examination, projected against the thorough and accurate record of effective prior examinations, contributes substantially to the acuity and significance of the health-control program. The essential elements are the history, physical examination, and laboratory studies.

A complete and detailed medical and occupational history is fundamental. Disease which might have developed since the last examination should be diligently searched for. Of specific interest will be conditions which have had or are likely to have radiation therapy, or which could be confused with radiation injury. Adequate time must be allotted to allow careful review of interim exposures to occupational hazards, particularly radiation. Exposures to diagnostic and therapeutic X-rays, changes in job status and responsibility, and alteration in attitudes or emotional patterns are important. Opportunities to discuss apprehensions or misinformation about radiation exposure and possible biological effects will arise and should be welcomed. Changes in family and marital status should be recorded.

A complete physical examination is usually regarded as essential. In certain circumstances it may be desirable to use limited specialized examinations—usually scheduled at more frequent intervals than is practical for full-scale medical examinations. Hearing, vision, physical capabilities, and temperament are evaluated in relation to the specific job requirements and hazards. The absence of findings that could result from overexposure should be recorded even though radiation monitoring has indicated that the individual's exposures have been below the permissible level. Laboratory studies should include routine urinalysis and complete blood count. The addition of other procedures such as audiometry, vision tests, electrocardiography, and blood and urine biochemistry will be determined by the individual's medical status or environmental nonradiation hazards.

While **diagnostic radiologic procedures** are ordinarily excluded in the computation of the cumulated permissible dose—"exposure of any part of the body to X-rays resulting from ordinary medical diagnostic procedures shall be assumed to have no

effect on the radiation tolerance status of the person concerned, provided no contributory accidental or emergency exposure . . . has occurred within the previous three months" (NBS Handbook 59, p. 73, Sept. 24, 1954)—special consideration should be given to limiting the radiation from this source to persons occupationally exposed. The risk of adding to the cumulated dose must be balanced against the potential value of the medical information needed. Particularly, the radiologic procedures which are to be used in the periodic examination should be selected for their probable high yield of pertinent information, and given with a minimum of radiation exposure. For example, chest radiographs may not be indicated on an annual basis for young healthy adults in a population relatively free from pulmonary disease or where there is no occupational pulmonary hazard but may be justified with increasing frequency after age forty to detect early pulmonary malignancy and heart disease. The standard 14- by 17-in. chest radiograph involves significantly less radiation exposure for the subject than does the photoroentgen technique and hence is preferable for radiation workers.

The general principles which should govern all diagnostic radiologic procedures have special meaning in relation to examinations of persons with occupational exposure to radiation: (1) the limitation to those tests which have definite medical indications and reasonable probable yield of pertinent information; (2) the reduction of the radiation exposure by limiting field size, adequate filtration, and local shielding; and (3) the availability of reasonable estimates of the radiation dose delivered per examination in the particular installation.

Special test procedures such as examination of the blood for specific abnormal blood-cell morphology, examination of excreta for radioactivity, or external body counting for assessment of internally deposited radioisotopes can be useful additions to the periodic medical examination or may be conducted at more frequent intervals depending on the type and magnitude of the person's exposure. The place of these procedures in radiation control is similar to that of special tests in the control of chemical hazards: for example, the determination of urinary coproporphyrin excretion, of basophilic stipple-cell counts, and of urinary and blood lead in persons exposed to a lead hazard.

While hematological procedures are an essential part of the preplacement and periodic examinations, **blood counts** are no longer recommended as a monitoring method in radiation protection by the NCRP:*

1. Provided that radiation monitoring of personnel, and where applicable of sites, is carried out by instruments (film badges, pocket meters, etc.) in all circumstances involving potential exposure to penetrating ionizing radiations, blood counts should no longer be required as a *method of monitoring*.
2. Blood counts as a part of pre-employment, interval, and terminal examination are good medical practice—to be done at the discretion of the medical officer in charge—but not as a part of a monitoring service.
3. Blood counts are a necessary part of the medical examination of anyone over-exposed to penetrating ionizing radiations.

For levels of radiation below or even slightly exceeding the maximum permissible dose, there is no medical test or procedure which is sufficiently sensitive or specific

* Blood Counts, Statement of the National Committee on Radiation Protection, *Radiology*, vol. 63, p. 428, 1954.

in identifying the earliest biological effects of radiation to justify routine application as a biological monitoring system. Such special hematological examinations as the measure of the increased incidence of lymphocytes with bilobed nuclei or the increased incidence of neutral red granules in lymphocytes show promise as sensitive indicators of radiation exposure but require further definition and validation.

At radiation levels moderately exceeding the maximum permissible dose, certain tissue changes may be encountered. Although the identification of these effects is not suitable for biological monitoring for control purposes, it is useful in evaluating the extent or severity of the excessive radiation exposure. One of the earliest tissue changes is an alteration in the numbers and morphology of the lymphocytes [Cronkite, E. P., *The Hematology of Ionizing Radiation* in Behren, Charles F. (ed.), "Atomic Medicine," 2d ed., chap. 8, p. 125, Baltimore, 1953]. Following chronic irradiation to the hands, a flattening and obliteration of the ridge detail of the fingertip skin may occur with a frequency and severity which are proportional to the duration of exposure (Nickson, M., "A Study of the Hands of Radiologists," MDCC-1309, 1947). Radiologic abnormalities of the bones in some persons with radium deposits of long duration have been described (Looney, W. B., *Late Skeletal Roentgenographic, Histopathological, Autoradiographic and Radiochemical Findings Following Radium Deposition*, *Am. J. Roentgenol. Radium Therapy*, vol. 75, p. 559, 1956). Morphologic changes in the hair roots of the scalp appear after acute exposures of 100 rems or more (Van Scott, E. J., and R. P. Reinertson, *Detection of Radiation Effects on Hair Roots of the Human Scalp*, *J. Invest. Dermatol.*, vol. 29, p. 205, 1957). At present too little information is available on the relationship of such findings to the development of later more serious radiation effects to give them high prognostic value.

Procedures for estimating the body content of radioactive materials are essential in protecting workers exposed to the risk of internal deposition of radioisotopes. Environmental monitoring cannot be relied upon as the sole control procedure in such circumstances. The use of **bio-assay procedures** as a method of personnel control was officially advised as necessary by the NCRP in 1941 in the case of workers using radioactive luminous compounds ("Safe Handling of Radioactive Luminous Compound," NBS Handbook H27, 1941). The application of these procedures has been extended to other radionuclides. ("Safe Handling of Radioactive Isotopes," NBS Handbook, p. 9, 1949; "Radiological Monitoring Methods and Instruments," NBS Handbook 51, p. 20, 1952.) The New York State Sanitary Code requires annual examination of all persons who might repeatedly ingest or inhale radioactive materials in concentrations that exceed one-quarter of the maximum permissible concentrations in air and water and states that such examinations shall include, according to the physician's judgment, pertinent laboratory examinations such as urinalysis, expired-air analysis, and other laboratory tests to determine the presence of radioactive material in the body (New York State Department of Health, *The Sanitary Code*, chap. XVI, Regulation 7, 1955). These recommendations express indirectly but clearly the need for and the usefulness of such procedures.

The application of these techniques requires considerable technical competence in radiochemical and physical methods. More importantly, they need detailed understanding of the metabolism of specific radioisotopes and of the relationship of the bio-assay results to the potentialities of organic radiation damage. The complex

variables involved in absorption, distribution, retention, and excretion of the radioisotopes must be evaluated and in turn related to the values for maximum permissible amounts of specific radioisotopes in the human body ("Maximum Permissible Amounts of Radioisotopes in the Human Body and Maximum Permissible Concentrations in the Air and Water," NBS Handbook 52, 1953).

The type or types of bio-assay will be determined by the specific radionuclides, by the character of available test procedures, and by the magnitude and temporal relations of absorption and sampling. The frequency of such procedures must be related to the type and severity of exposure. For example, in the routine operations in the processing of uranium compounds, urinalyses for uranium may be performed at regular intervals of 1 week, 1 month, or every 6 months, depending on the risk for exceeding permissible levels of absorption. In the start-up of a new operation, or in any unusual change in operations, samples may be collected and analyzed daily, or even every few hours to aid in localizing the stages in the operation associated with the greater exposure.

The amount of uranium contained in a "spot" sample of urine collected during an exposure period is difficultly related to the real body burden of uranium, since it may reflect chiefly the recently absorbed and transient uranium. A sample collected after several days' freedom from exposure is more indicative of the retained component. The relative acuity, accuracy, and facility of urine uranium determinations, and the accumulated knowledge relating uranium excretion to retention, make this procedure an excellent example of a practical, reliable, and simple routine control method.

Reasonable procedures for estimating excretion and body content, in relation to the control of occupational exposures, have been reported for a number of important radioisotopes, including radium, plutonium, polonium, strontium, cesium, thorium, and iodine. For example, with radium, several different kinds of analysis have been and are being used. The determination of the radon content of the expired air has been used to indicate the magnitude of radium absorption. The determination of radium in the urine and feces is another method of measuring radium absorption. More recently, the development of extremely sensitive whole-body counting devices has permitted the in vivo determination and localization of quantities considerably smaller than the permissible body burden.

In any medical control program, provisions for management of **overexposures in emergencies** must be made. Their extent depends upon the potentialities for accident. In an installation performing only industrial radiography, little need be done in this regard. In installations such as fuel-processing plants or power reactors where high-level contamination of persons or wounds could occur, more elaborate provisions would be necessary. In addition to cooperating with other interested parties in developing an over-all disaster plan (fire control, respirator supply and control, evacuation, provision for increased monitoring, etc.) medical personnel must see that they are provided with adequate stand-by facilities for handling decontamination of persons and for treating contaminated wounds. Floor plans of treatment and decontamination facilities, equipment, and special procedures designed for coping with such problems have been described (Finkel, A., and E. A. Hathaway, *Medical Care of Wounds Contaminated with Radioactive Materials*, *J. Am. Med. Assoc.*, vol. 161, p. 121, 1956).

In any case in which there has been overexposure, internal or external, every effort must be made to establish the magnitude of exposure and to describe fully and serially any effects. These should include a record of all the relevant details of the accident, an evaluation of personnel monitoring devices, the monitoring of skin and wounds. In some cases it may be helpful to utilize area monitoring during a reproduction of the accident. The determination of induced radioactivity in body fluids and metal-filled teeth has been utilized in neutron exposures. The measurement of radioactivity in nasal smears and sputum is useful in individuals who have inhaled radioisotopes. Repeated clinical and laboratory observations such as hematologic tests; bio-assays on blood, urine, and feces; and an examination for skin and hair-root reactions should be energetically pursued to guide in the management of the patient and to add to the general fund of knowledge.

The **treatment** for persons receiving **excessive doses** of external **radiation** is supportive, with no specific therapy of practical value. The accepted principles include: rest, asepsis, transfusions, antibiotics, maintenance of fluid and electrolyte balance, and adequate nutrition.

In cases of excessive exposure to unsealed radioisotopes, prompt action may decrease absorption or hasten elimination. Immediate and energetic decontamination of the skin, accessible body orifices, and wounds should be carried out. Where there has been possible ingestion of radioisotopes, evacuation of stomach contents and catharsis may be indicated. For example, magnesium sulfate has been suggested for gastric lavage and daily catharsis in radium exposures. Administration of large amounts of water will reduce the biological half-life of absorbed tritium. The use of complexing or chelating agents, such as zirconium citrate or calcium ethylene diamine tetracetic acid, administered intravenously, has been shown experimentally to increase excretion and decrease deposition of certain radioisotopes. The clinical experience is limited but promising. Further information is needed before this type of procedure can be recommended as routine therapy. No practical methods exist for removing more than small fractions of radioisotopes deposited in bone. With the limited effectiveness of procedures for increasing excretion or elimination of many isotopes, greater emphasis must be placed on techniques for prevention of absorption.

After significant technical overexposure or after recovery from the acute effects of actual overexposure, decision about future potential exposure to radiation for that individual should be made by experts qualified in medical radiology, radiobiology, and radiological physics. The final decision in an individual case will depend upon the balance of these factors:

1. The nature of the accident: severity, external vs. internal exposure.
2. The characteristics of the individual: health, age, sex, intelligence, training, interests.
3. The nature of the job: chance for repeat overexposure, general level of chronic exposure, availability of modified occupation.
4. The regulations governing radiation exposure: governmental, local industry.

MEDICAL RECORDS

Medical records serve multiple purposes. An individual's record gives a cumulative account of his health from the time of employment to discharge and thus serves in the promotion and maintenance of his personal health. The medical records of groups of people in an occupational population—a specific exposure group, a department, a plant, or a company—furnish data for the periodic assessment of the health of the group. They are the raw material for evaluating the activities of the health-protection services. In an increasing number of states, the maintenance of records of medical examinations for individuals with occupational radiation exposure is required by law. But even in the absence of such statutory requirements, good medical records are essential in assuring the safety of radiation operations and in protecting both employee and employer in potential medicolegal problems.

To fulfill these purposes records must adequately identify the individual, be accurate and complete, and inspire confidence. Precision in individual identification is more likely when full name, date of birth, and some permanent personal identifying number are used. The personal number may be a group-insurance number permanently assigned at the time of employment with the company, a personal record number, or a social security number. Identifying numbers which are changed with inter- or intradepartment transfers, such as payroll numbers, should not be used. Order, neatness, and legibility encourage confidence in the accuracy of the record. The originator of each entry in the record must be clearly identified. The importance of recording both positive and negative data cannot be overemphasized. The absence of accurate data may handicap the interpretation of subsequent findings to the disadvantage of both the employee and employer, especially in medicolegal problems. With respect to radiation exposure, the record of negative or insignificant environmental and bio-assay values is of particular importance.

The important medical data for personnel control in radiation protection have been outlined in the discussion of the selection of personnel, the periodic examination, and special test procedures. Quantitative and descriptive information on radiation exposures such as personnel monitoring, area monitoring, and bio-assay determinations must be available in the medical record. In many situations the **reports of film-badge and dosimeter readings** and area surveys are maintained in a file separate from the medical records (for example, by the health-physics or industrial-hygiene services). It will therefore be necessary to summarize periodically the information concerning an individual's exposure and enter it in his personal medical record. The frequency of this periodic assembly, and the simultaneous appraisal of monitoring and medical records will vary, but it ought not be less often than once a year. In any case where there has been overexposure, radiation therapy, or appreciable diagnostic radiation, the best possible estimate of the dose should be recorded and evaluated in light of the previous total exposure. Although radiation from medical procedures is specifically excluded in the computation of the occupational permissible dose by the NCRP, prudence dictates the consideration of radiation exposure from all sources in evaluating the hazard of continuing employment with radiation. At the time of discharge a complete summary of the employee's

radiation exposure should be made. If his subsequent occupation involves radiation exposure, information from his medical and exposure records may be requested. In case medicolegal questions arise, the complete record will be readily available.

The **medical-record form** will vary in size, organization, and detail in different situations. No single form has been devised that is universally applicable or acceptable. In general, the radiation-exposure data will be entered on a special form or sheet added to the medical record, summarized in a place provided on a periodic-examination form, or entered in chronological order in a narrative record of medical findings and events. Summarization and comparison are facilitated by the first two arrangements. In many companies the number of records and mass of data will be large enough to require mechanical handling through the use of a punch-card system. The simpler of these systems, such as the hand-operated key-sort method, can be used where the number of individuals and events is limited. Where larger numbers are involved, machine systems are needed for effective handling. The punch cards themselves do not constitute adequate records but save time and effort in compiling and analyzing the exposure and health information.

"Suitable filing equipment and training of personnel should be provided for the safekeeping and confidential maintenance of all medical records in the exclusive custody and control of medical personnel." (Council on Industrial Health, Guiding Principles of Medical Examinations in Industry, *J. Am. Med. Assoc.*, vol. 161, p. 975, 1956.) The confidential nature of medical records should be closely guarded and information released only on written consent of the employee. Consent should be obtained before information is supplied to other physicians, insurance, or health agencies. Governmental health authorities, courts, and workmen's compensation commissions may be given information where required by law or official orders. Generalizations regarding a person's health, in terms of his capabilities, can be given ethically to the employer for the purpose of job placement or adjustment of working conditions.

Policies on the **retention of medical records** vary. The usual practice is to retain an individual's record throughout his employment and for some finite period after his discharge or death. Since the latent period between radiation exposure and harmful effect may be many years, there is justification for retaining medical records of exposed persons for long periods. Increasingly, governmental agencies are stipulating retention requirements for medical and exposure records. For example, the New York State Sanitary Code requires records to be kept for the length of employment plus 5 years, or for 2 years after death, or until specific instructions are given by the State Commissioner of Health (New York State Department of Health, The Sanitary Code, chap. XVI, Regulation 7, 1955). The New York State Department of Labor, in effect, provides for longer retention of records (medical as well as monitoring and survey) by not permitting destruction of records without the consent of the Commissioner of Labor, by requiring records to be available to attending physicians of anyone who may have had radiation exposure and to the Commissioner, and by empowering the Commissioner to receive and keep or destroy records (State of New York Department of Labor, The Industrial Code, Rule 38, sec. 38-8, 1955). Where no official retention requirement exists, it is recommended that records be maintained at least 2 years after death, with ultimate disposal as advised by the health agency having primary jurisdiction in the area.

MEDICOLEGAL CONSIDERATIONS

The complex and diverse characteristics of radiation exposure and injury have created, and will create in increasing frequency and importance, many and difficult **medicolegal problems**. The range of injurious effects, from a superficial and localized tissue change of minor significance, to a potential shortening of life span or genetic changes involving one or more future generations, presents a great potential for legal action.

Even in limiting this discussion to occupational relationships, the varied pattern of **workmen's compensation laws** in the various states prevents a generalization as to the type and extent of coverage for radiation injury. In a few states, the "all-inclusive" character of the law clearly includes the compensation of individuals injured in the course of their work by radiation. In other states, an "accident" involving radiation injury might be covered under the law, while an injury from repeated and prolonged exposure might be excluded from such coverage unless "radiation" was an expressly designated category in an occupational-disease statute. An injured workman would then have recourse to a common-law action, with such factors as negligence on the part of the employer or contributing negligence by the employee weighing heavily in the result.

The **diagnosis** of a severe, **acute radiation injury**, with the confirmation of the adequate environmental factor, is relatively simple. But even where these factors are recognized and accepted, the estimation of sequelae in terms of disability, and ultimately monetary compensation can be complicated and difficult. The diagnosis of radiation injury or alleged radiation injury, of lesser degree, particularly where the environmental factors are less clearly defined, can be extremely confused and uncertain. Injury from radiation is not pathognomonic, that is, not so exclusively characteristic or distinctive as to identify radiation as the specific cause of the injury to the exclusion of other reasonable causes. Such diseases as cancer, leukemia, aplastic anemia, and such conditions as shortening of the span of life and genetic mutations can be caused by excessive radiation exposure. It is unlikely, however, that radiation, and certainly not occupationally induced radiation, is a major etiologic factor in their present incidence in the general population. There is no practical or reasonable method for assessing the contribution in a particular instance of an excessive occupational exposure to radiation in the case of a "radiation-associated" disease. Until better diagnostic criteria have been developed, the considered judgment of competent and experienced clinicians in the field of radiation effects will have to be accepted as the best answer.

With an increasingly liberal interpretation of the compensation laws with respect to "the benefit of the doubt" in favor of the plaintiff, reliable and complete environmental data will be required to defend successfully against an alleged excessive exposure to radiation. The reports as to environmental, bio-assay, and medical findings must be complete, quantitative, reliable, and verified. The summarization at regular intervals, preferably not less than annually, should be done with the best possible estimate of the total exposure. Negative data should be reported in terms of "not more than" or "less than." While this reporting of "negative data" may appear to be needlessly wasteful of effort and money, the result will be justified by

the security from false claims and by the potential contribution to a better understanding of the effects of radiation.

General References

- Guiding Principles of Medical Examinations in Industry, *J. Am. Med. Assoc.*, vol. 161, pp. 975-978, July 7, 1956.
- Mittelman, B., *et al.*, The Cornell Indices, The Cornell Word Form, *Ann. N.Y. Acad. Sci.*, vol. 46, p. 573, 1946.
- NBS Handbook 59, p. 59, Sept. 24, 1954.
- NBS Handbook 59, p. 73, Sept. 24, 1954.
- Blood Counts, Statement of the National Committee on Radiation Protection, *Radiology*, vol. 63, p. 428, 1954.
- Cronkite, E. P., The Hematology of Ionizing Radiation, in Behrens, Charles F. (ed.), "Atomic Medicine," 2d ed., chap. 8, p. 125, Baltimore, 1953.
- Nickson, M., "A Study of the Hands of Radiologists," MDDC-1309, 1947.
- Looney, W. B., Late Skeletal Roentgenographic, Histopathological, Autoradiographic and Radiochemical Findings Following Radium Deposition, *Am. J. Roentgenol. Radium Therapy*, vol. 75, p. 559, 1956.
- Van Scott, E. J., and R. P. Reinertson, Detection of Radiation Effects on Hair Roots of the Human Scalp, *J. Invest. Dermatol.*, vol. 29, p. 205, 1957.
- "Safe Handling of Radioactive Luminous Compound," NBS Handbook H27, 1941.
- "Safe Handling of Radioactive Isotopes," NBS Handbook 42, p. 9, 1949.
- "Radiological Monitoring Methods and Instruments," NBS Handbook 51, p. 20, 1952.
- New York State Department of Health, The Sanitary Code, chap. XVI, Regulation 7, 1955.
- Maximum Permissible Amounts of Radioisotopes in the Human Body and Maximum Permissible Concentrations in the Air and Water, NBS Handbook 52, 1953.
- Finkel, A., and E. A. Hathaway, Medical Care of Wounds Contaminated with Radioactive Materials, *J. Am. Med. Assoc.*, vol. 161, p. 121, 1956.
- Council on Industrial Health, Guiding Principles of Medical Examinations in Industry, *J. Am. Med. Assoc.*, vol. 161, p. 975, 1956.
- State of New York Department of Labor, The Industrial Code Rule 38, sec. 38-8, 1955.

INDEX

- Absorbed dose, **2-9, 3-18, 14-2, 14-3**
 - from point source, **14-3**
 - X- or gamma radiation, **14-3**
- Absorption, **2-2**
 - photoelectric, **10-16**
- Absorption coefficient, **2-2, 6-49, 6-50**
- Absorption curve, **2-2**
- Acceleration, **2-3**
- Accelerator, **6-33 to 6-44, 12-24**
 - high-energy, **6-59, 6-60**
 - induction (betatron), **2-6**
 - linear, **2-14, 6-43, 6-71**
 - particle, **6-33 to 6-60**
 - resonance, **6-34**
 - safety, **12-20**
 - shielding, **6-54 to 6-59**
- Access, **17-28**
- Access hatch, **17-29**
- Access port, **17-29**
- Accident, **11-2**
 - in reactor, **11-34, 12-16**
- Accident prevention, **11-3**
- Actinium determination, **16-8**
- Actinium series, **6-4**
- Active, definition, **2-3**
- Active deposit, definition, **2-3**
- Acute exposure, **14-16**
- Adsorption, on clay, **21-23 to 21-27**
 - surface, **21-14**
- AEC (*see* Atomic Energy Commission)
- Aerosol, **20-2, 22-2, 22-25, 22-32**
 - generation, **20-14**
- Air, **8-31**
 - radioactivity, **4-12**
- Air cleaning, **22-6**
- Air-density correction, **1-29**
- Air dose, **2-9**
- Air ejector, **20-21**
- Air entry, **22-7**
- Air filtration, **20-3**
- Air-flow control, **22-7**
- Air pollution, **11-12, 22-2 to 22-45**
 - control methods, **22-5**
 - sources of, **22-2**
- Air sample, analysis, **11-6**
- Air sampling, **11-4, 12-8, 20-3, 20-29**
- Aircraft, **3-65**
- Aircraft-wing-tip carrier, **17-34**
- Alpha decay, **5-8**
- Alpha emission, **5-6**
- Alpha emitter, **2-3, 13-18**
- Alpha particle (ray), **2-3, 10-16, 10-35, 17-6**
- Alpha source, **13-21**
- Aluminum, **8-11, 8-33**
- American Conference of Governmental Industrial Hygienists (ACGIH), **3-34**
- American Standards Association (ASA), **3-31 to 3-33**
- Americium determination, **16-9**
- Amplifier, **10-11, 10-26**
- Anemia, **2-3, 19-10**
 - aplastic, **19-7, 19-10**
- Angstrom, **2-3**
- Angular distribution of X-rays, **6-48**
- Ankle, **3-14**
- Annihilation, **2-3, 5-11**
- Anode, **6-62**
- Anthracene, **10-19**
- Area monitoring, **2-4, 2-15, 10-58**
- Area survey, **12-4**
- Argon, **8-15, 10-35, 10-48, 11-29**
- Argon-41, **14-9**
- Artificial radioactivity, **2-4**
- ASA (*see* American Standards Association)
- Atmosphere, properties of, **1-30**
- Atmospheric dilution, **22-26**
- Atmospheric dispersion, **22-32**
- Atom, **2-4**
- Atomic energy, **2-4**
- Atomic Energy Act of 1954, **3-35**
- Atomic Energy Commission (AEC), **3-35**
 - Radiation Protection Standards, **3-36**
 - regulations, **3-35 to 3-38**
 - rules of practice, **3-36**
- Atomic number, **2-4**
 - index of, **1-3**
- Atomic weight, **2-4**
- Attenuation, **2-4, 8-2**
 - broad-beam, **7-9, 8-2**
 - narrow-beam, **7-6**
- Attenuation coefficient, **7-5, 7-7**
- Attenuation data, **8-2 to 8-69**
 - theoretical, **8-32**
- Attenuation laws, **7-5**
- Autoradiograph, **2-5**
- Autoradiography, particle, **20-18**
- Avalanche, **10-34, 10-37**
- Average life, **2-5**
- Background, **2-5, 10-39, 10-44, 11-13**
 - natural, **4-1 to 4-18**
- Background radiation, **2-17**
- Backscatter, **2-5, 10-68**
- Badge, film, **2-11, 12-11**

- Barn, **2-5**
- Barricade, brick, **17-17**
 - reach-around, **17-16**
- Barrier, **13-14, 17-15**
 - primary, **8-43, 13-4**
 - secondary, **8-46, 8-53, 13-4**
- X-ray, **8-51**
 - multimegavolt, **8-64, 8-65**
- Barrier shield, **9-4**
- Beam, **2-5**
 - useful, **13-3**
- Beam-defining devices, **13-16**
- Bench, **9-4, 18-11**
- Beryllium, **8-5, 11-26, 22-4**
- Beryllium window, **13-12**
- Beta barrier, **17-16**
- Beta decay, **2-5, 5-8, 5-10**
- Beta disintegration, **2-5**
- Beta emitter, **2-5, 13-18**
- Beta monitoring, **18-13, 18-14**
- Beta particle (*see* Beta radiation)
- Beta radiation (ray), **2-5, 2-6, 7-2, 10-35, 10-54**
 - average energy, **14-16**
 - coefficient of absorption, **14-6**
 - energy, **7-4**
 - film response, **10-69**
 - mixed, **10-70**
 - monitoring, **10-70**
 - range, **14-6**
 - spectrum, **5-8**
- Beta-ray applicator, **13-20**
- Beta-ray source, forms, **11-42**
 - sealed, **11-42**
 - uses, **11-42**
- Beta spectrum, **5-8**
- Betatron, **6-37, 6-65, 6-70**
 - induction accelerator, **2-6**
- Betatron-synchrotron radiation, **3-10**
- Bev, **2-6**
- Bevatron, **6-42**
- Bio-assay, **12-4, 12-15, 23-9**
- Biochemical process, **21-33**
- Biological effect, **19-2**
- Biological effectiveness, **2-6**
 - half-life, **2-6**
 - high-energy particle, **13-15**
 - relative (*see* Relative biological effectiveness)
- Biological effects of neutrons, **13-15**
- Biological processes, **21-33**
- Biological sampling, **11-9, 11-11**
- Blood change, **19-6**
- Blood count, **2-9, 19-10, 23-8**
- Blood-forming organ change, **19-7**
- Body, human, radioactivity of, **4-15**
 - radium in, **4-16**
 - total radiation, **19-2**
- Body burden, **3-8, 3-18, 3-28, 11-10, 14-13, 14-14**
- Body excretions, **14-17**
- Bone-marrow death, **19-3**
- Bone-marrow injection, **19-4**
- Bone seeker, **2-6**
- Boron chamber, **2-6**
- Boron counter, **2-6**
- Bottle capper, **17-9**
- Boxcar, **17-34**
- Bragg-Gray theorem, **10-4**
- Breathing zone, **20-25**
- Breeder, **6-77**
- Bremsstrahlung, **2-6, 5-11, 6-48, 7-2**
- Brick shielding, **1-37, 17-17**
- Broad-beam attenuation, **7-9, 8-2**
- Bromine, **10-48**
- Build-up, **2-7, 8-34 to 8-36**
- Build-up factor, **7-10, 8-3, 8-33**
- Building decontamination, general, **18-19**
- Building survey, **12-4**
- Burial, costs, **21-53**
 - land, **21-45**
 - sea, **21-31**
- By-product material, **2-7**
 - (*See also* Isotope)
- By-product material license, **3-37**
- Bypass air, **22-18**
- Cadaver, handling, **3-10, 3-11**
- Cadmium foil, **10-72**
- Calcium, **8-17**
- Calcium phosphate, **8-30**
- Calibration, **13-17**
 - of counter, **10-38**
 - of film, **10-63**
- Canal, **17-31**
- Cancer, **19-14**
- Capture, **2-7**
- Carbon, **8-6**
- Carcinogenesis, **2-7, 19-9, 19-14**
- Carcinogenic, definition, **2-7**
- Carcinogenic effect, **19-7**
- Cascade, **10-34**
- Cascade impactor, **20-20**
- Cataract, **19-11**
- Cathode, **6-62**
- Cathode-ray tube, **6-71**
- Cave, **22-5**
 - junior, **17-22, 22-22**
- Cell, process, **17-24**
- Central-nervous-system death, **19-3**
- Cesium-137, **3-9, 11-40**
- Change area, **9-8**
- Characteristic radiation, **2-17**
- Charged-particle reaction, **7-12**
- Chemical dosimetry, **10-76 to 10-78**
- Chemical laboratory, **11-46**
- Chemical precipitation, **21-13**
- Chemical procedures, **16-4**
- Chemistry, **16-2**
 - (*See also* specific elements)
- Chlorine, **10-48**
- Chlorophenol red indicator, **10-77**
- Civil Aeronautics Board, **3-63**
- Civil Air Regulations, **3-63**
- Clay, **21-20**
 - adsorption on, **21-23 to 21-27**
 - colloidal, **21-27**
 - exchange properties, **21-23**
 - montmorillonite, **21-27**
 - porosity, **21-27**

- Cleaning agent, liquid, **18-3**
- Cleaning device, **22-8**
- Cleaning equipment, **22-45**
- Close shields, **17-11**
- Closed cell, **17-6**
- Closed-cell equipment, **17-23**
- Clothing, **18-12**
 - decontamination, **18-19**
 - protective, **11-23, 11-49, 12-8**
- Cobalt-60, **3-9, 11-38, 11-39, 13-19**
- Cockcroft-Walton circuit, **6-33**
- Code, **11-14**
 - industrial hygiene, **3-34**
 - state, **11-2**
- Collector, inertial, **22-32**
- Collimator, **2-7**
- Colloidal suspension, **13-19, 13-22, 13-24, 13-25**
- Color, radiation hazard, **3-32**
- Complexing agents, **21-17**
- Compton collision, **10-15**
- Compton process, **7-7**
- Concentration in contamination control, **17-4**
- Concrete, **8-29, 17-4**
- Concrete shielding, **1-36, 1-37**
- Condensation, **11-5**
- Congenital abnormalities, **23-3**
- Constant-potential circuit, **6-69**
- Constants, fundamental, **1-5**
- Construction materials, **17-24**
- Contact therapy, **13-12**
- Containment, in contamination control, **17-4**
 - and decay, **22-30**
- Contaminability, **18-3**
 - of surface, **18-22**
- Contaminant, **22-6**
 - air-borne, **20-2, 22-2**
 - gaseous, **20-12**
 - liquid-drop, **18-3**
- Contaminated combustibles, **21-54**
- Contaminated metal, **21-58**
- Contamination, **2-7, 3-6, 11-47, 13-26, 18-4, 18-22**
 - loose, **18-14**
 - sources of, **18-2, 18-3**
 - surface, **11-21, 12-9, 18-2 to 18-23**
 - total, **18-14**
- Contamination control, **18-9**
- Contamination control equipment, **17-13**
- Continuous sampling, **11-6**
- Control film, **10-55**
- Controlled area, **3-14**
- Conversion coefficient, **5-13**
- Converter, **6-77**
- Coolant (*see* Reactor coolant)
- Cooling air, **22-5**
- Cooling fluid, **21-9**
- Cooling water, **22-5**
- Copper, **8-19**
- Coprecipitation, **16-4**
- Cosmic radiation, **4-3 to 4-6**
 - data, **4-6**
 - intensity, **4-4**
 - primary, **4-3**
- Cosmic radiation, secondary, **4-4**
 - showers, **4-4**
- Cosmotron, **6-42**
- Coulomb factor, **5-10**
- Count, **2-7, 10-37, 10-44**
- Count-rate meter, **2-7**
- Count-rate nomograph, **1-40 to 1-42**
- Counter, **2-7, 10-33**
 - Geiger, **10-33 to 10-52**
 - halogen, **10-48**
 - neutron, **10-40**
 - proportional, **10-33 to 10-41**
 - scintillation, **10-14 to 10-32**
- Counter tube, **10-37**
- Counting, alpha-track, **20-19**
 - beta, **16-3**
 - gamma, **16-3**
- Counting apparatus (equipment), **11-6, 13-3, 16-3**
- Counting-room location, **16-2**
- Crib, **21-41**
- Critical, definition, **2-7**
- Critical mass, **11-24**
- Critical organ, **3-14, 3-19, 14-13, 14-18, 19-12**
- Criticality, **2-8, 15-2 to 15-11**
- Cross section, **2-8, 6-74, 7-6**
- Cumulative dose, **2-9**
- Curie, **2-8**
- Cutie Pie, **10-10**
- Cyclotron, **2-8, 6-33, 6-34, 6-36**
- Daughter, definition, **2-8**
- DBP, **16-21**
- Dead time, **10-36, 10-44, 10-46**
- Death from radiation, **19-3**
- Decay, **2-8, 5-8**
- Decay chain, **5-5**
- Decay rate, **5-2, 5-4**
- Decontaminability, **18-3, 18-22**
- Decontamination, **2-8, 11-30, 11-32, 18-15**
 - approach to control, **17-3**
 - building material, **18-16**
 - clothing, **18-19**
 - liquid, **18-7**
 - personnel, **18-19**
 - stepwise, **18-7**
 - surface, **18-2 to 18-23**
- Decontamination factor, **18-8**
- Decontamination index, **18-8, 18-10**
- Decontamination reagents, **18-15**
- Decontamination test, **18-8**
- Deep disposal, land, **21-45**
 - sea, **21-39 to 21-41**
- Deep therapy, **13-13**
- Delayed neutron, **6-76**
- Densitometer, **2-8**
- Density, **2-8, 15-3, 15-7**
- Deposition, **22-32**
- Depth dose, **2-9, 2-10**
- Detection of radiation, **10-2 to 10-84**
(*See also* Measurement of radiation)
- Deuteron, **2-9**
- Diagnosis, radiologic procedures, **23-7**
(*See also* X-ray diagnosis)
- Dibutylphosphoric acid, **16-21**

- Diffusion in decontamination, **18-7**
- Dilution in contamination control, **17-3**
- Diethylphthalate, **22-25**
- Direct viewing, **17-6**
- Discharge, and dilution (waste), **21-34**
 - from hospitals, **21-34**
- Discomposition, definition, **2-9**
- Discriminating circuit, **10-39**
- Discrimination, **10-14**
- Disintegration, definition, **2-9**
 - of nuclei, **6-34**
- Disintegration constant, **5-10**
- Dispersal (contamination control), **17-3**
- Disposal (*see* Waste disposal)
- Distance control, **17-7**
- Dominant radiation, **6-55**
- Donut, **6-65, 6-70**
- DOP, **22-25**
- Dosage (*see* Dose)
- Dose, absorbed, **3-18, 14-2, 14-3**
 - definition of terms, **2-9, 2-10**
 - doubling, **19-16**
 - exposure, **14-2**
 - integral, **13-16**
 - internal sources, **14-10**
 - maximum permissible (*see* Maximum permissible dose)
 - median lethal (LD-50), **2-10, 19-3**
 - natural radiation, **4-17**
 - tissue, **2-10**
 - tolerance, **2-10**
- Dose meter, **10-8**
- Dose-rate meter, **10-9**
- Dose registry, **3-5**
- Dosimeter, chemical, **10-76**
 - coloration, **10-81**
 - condenser, **10-10**
 - pocket, **10-9**
 - definition, **2-16**
 - radiophotoluminescent, **10-81**
 - solid, **10-80**
 - stimulated luminescence, **10-84**
 - thermoluminescence, **10-84**
- Dosimetry, **10-2 to 10-84**
 - chemical, **10-76**
 - optical, **10-79**
 - photographic, **10-53**
- Doubling dose, **19-16**
- Drain line, **11-15**
- Drain system, **11-15, 11-16**
- Drift velocity, **10-37**
- Drosophila, **19-16**
- Dry box, **9-6, 11-47, 22-21**
- Dry mines, **21-45**
- Duct, **22-7**
- Dust, **11-18, 20-2, 22-2**
- Dust concentration, **11-22**

- Earth, crust of, radioactivity, **4-7**
- Effective cross section, **6-81**
- Effective half-life, **2-12, 14-15**
- Efficiency, **2-10, 10-44, 20-27**
- Effluent, liquid, **11-14**
 - stack, **22-45**
- Elastic scattering, **7-2, 7-11**
- Electrodeposition, **16-4, 16-11**
- Electrometer, **2-10**
- Electron, definition, **2-10**
- Electron accelerator, **6-52**
- Electron microscope, **6-72**
- Electron range, **7-3**
- Electron volt (ev), **2-10**
- Electronic collision, **14-6**
- Electronic equilibrium, **10-68**
- Electronic tube, **3-37**
- Electrons, conversion, **5-13**
 - energy loss, **7-4**
 - high-energy, **7-3**
 - monoenergetic, **7-3**
- Electroscope, **10-9, 10-10**
 - definition, **2-10**
- Electrostatic generator, **6-33**
 - Van de Graaff, **6-70**
- Electrostatic precipitator, **11-4, 20-6, 22-34**
- Elements, and atomic numbers, index of, **1-3**
 - half-lives, **1-4**
- Embryo, effects of radiation, **19-13**
- Emergency measures, **11-43**
- Emitter, internal, **13-24, 13-25**
- Emotional factors, **23-4**
- Enclosure, **22-7**
- Energy absorption coefficient, **7-9**
- Energy dependence, **2-11**
- Energy-dependence film, **10-64**
- Energy spectrometry, **10-18, 10-24**
- Entrainment, **21-32**
- Entry route, **11-3, 11-7**
- Epidermitis, **19-8**
- Epilation, **19-9**
- Epithelitis, **19-8**
- Epithermal neutron, **7-10**
- Equilibrium, **2-11, 5-6**
 - secular, **2-11**
 - transient, **2-11**
- Erythema, **19-8**
- Erythema dose, **3-3**
- ev (electron volt), definition, **2-10**
- Evaporation, **21-27, 22-42**
 - costs, **21-30, 21-32**
 - vapor-compression, **21-30**
- Evaporator, natural-circulation, **21-31**
- Exempt quantity, **3-38**
- Exhaust, local, **22-6**
- Exhaust hood, **11-23, 22-7**
- Exit dose, **2-9**
- Expert, qualified, definition, **3-19**
- Exponential functions, values and logarithms of, **1-25 to 1-29**
- Exposure, acute, **14-16**
 - external, **11-38**
 - of man to background, **4-16**
 - nonoccupational, **19-7**
 - to several sources, **14-17**
 - (*See also* Radiation exposure)
- Exposure dose, **14-2**
 - from point source, **14-2**
- Exposure-time control, **17-7**
- External dose hazard, alpha sources, **14-5**
- External radiation, **11-3**
- External source, **13-24**

- Extraction with TTA, **16-7**
Extrapolation chamber, **10-12**
Eyes, effect of radiation, **19-11**
- Fall-out tray, **11-5, 20-11**
Fast neutron, **7-11, 10-72, 14-6**
Fermi plot, **5-10**
Fertility, **19-12, 19-13**
Fetus, effects of radiation, **19-13**
Filament, **6-62**
Film, **10-55, 13-3, 13-10**
 calibration, **10-63**
 commercial, **10-75**
 sensitivity, **10-56, 10-62, 10-70**
Film badge, **2-11, 12-11**
Film processing, **10-59, 10-60**
Film-processing apparatus, **10-61**
Film response, beta radiation, **10-69**
Film ring, **2-11**
Filter, **9-5, 11-48, 20-3**
 all-mineral, **22-37**
 contaminated, **22-45**
 dry, **22-36**
 efficiency, **20-4**
 high-efficiency, **22-32, 22-34**
 lead shot, glass fibers, **20-6**
 millipore, **20-6**
 sand and gravel, **22-38**
 wet, **22-41**
Filter indicator, **13-17**
Filter paper, asbestos and cellulose, **20-6**
 efficiency, **20-5**
Filtration, **13-11, 20-3**
 minimum, **13-4**
Filtration device, **11-4**
Finger-ridge pattern, **19-9**
Fire hazard, **11-22**
 uranium, **11-22**
Fission, **2-11, 11-26**
 (See also Criticality)
Fission process, **11-26**
Fission product (FP), **2-11, 6-77**
 contaminability, **18-6**
 release, **11-34**
Flooring, **18-11**
Flow counter, **2-11**
Flowmeter, **20-22**
Fluoroscope, **13-6**
 shoe-fitting, **3-34, 13-7**
Fluoroscopy, **6-63, 6-66**
Flux, **16-14**
E-M cyclotron, **2-21**
Focal spot, **6-62**
Focusing, magnetic, **6-59**
Fog, **20-2, 22-2**
Foil, intensifying, **10-67, 10-68**
Foot and ankle, **3-14**
Forced-circulation system, **21-29**
Forceps, **17-7**
Forearm, **3-14**
FP (see Fission product)
Fractionation, **19-2**
Fuel element, **22-5**
 spent, **11-33**
Fuel processing, **21-10, 21-11**
 Fume, **20-2, 22-2**
 Fume hood, radiochemical, **17-13**
 Fundamental constants, **1-5**
- Gamma emitter, **9-4, 13-18**
Gamma monitors, **12-23**
Gamma radiation, **2-12, 5-11, 6-46, 7-5 to 7-10**
 counter, **10-50**
 energy absorption, **7-9**
 source, **11-37**
Gamma shielding material, **17-25**
Gas, **22-2, 22-25**
 amplification, **2-12, 10-34**
 discharge, **10-42**
 ionization, **7-4**
 mixture, **20-2**
 multiplication, **10-6**
 noble, **22-44**
 polyatomic, **10-36**
 radioactive, **11-8, 22-7, 22-25**
 sampling, **20-12, 20-14**
 (See also specific gas)
Gas counter, **2-12**
Gas-filled tube, **10-44**
Gas-flow counter, **2-12**
Gaseous diffusion, **22-4**
Gastrointestinal tract, **14-15, 14-16, 19-3**
Geiger-Müller counter, **10-41 to 10-52**
Geiger-Müller region, **10-44**
Geiger-Müller threshold, **10-44**
Geiger-Nuttall law, **5-6, 5-8**
Gene, **2-12**
General air, **20-25**
General ventilation, **22-23, 22-24**
Generator, electrostatic, **6-33**
 Van de Graaff, **6-70**
 resonance, **13-13**
 Van de Graaff, **13-13**
Genetic effects, **3-5, 19-7**
Genetic mutation, **19-16**
Geologic factor, **21-7**
Geometry, **2-12**
Geometry factor, **20-27**
Giant resonance, **6-52**
Glass paper, **20-6**
Gloved box (glove box), **9-6, 17-14, 22-5, 22-21**
Gonads, **13-10**
Grab sample, **20-30**
Granulocytopenia, **19-3**
Grenz ray, **2-12, 6-65, 13-2**
Grinder, **22-12**
Gripping devices, pneumatic, **17-9**
Guard ring, **2-12, 10-6**
- H and D curve, **10-53**
Hack-saw enclosure, **22-10**
Half-life, **2-12**
 biological effectiveness, **2-6**
 effective, **2-12, 14-15**
 multiple, **1-43**
 of radioactive elements, **1-4**
Half-thickness, **2-13**

- Half-value layer, **8-47, 13-11**
 - multiple, **1-43**
- Halogen counter, **10-48**
- Hand and forearm, **3-14**
- Handling, **13-26**
 - equipment, **17-7, 18-12**
 - techniques, **17-6**
 - wet, **22-6**
- Handwork hood, **22-10**
- Hazard, relative, **14-14**
- Health examinations in industry, **23-7**
 - (See also Physical examination)
- Health physics, **2-13, 12-4**
- Heavy metal, **22-7**
- Helium, **10-48**
- High-energy electron, **7-3**
- High-energy particle, biological effects, **13-15**
- High-energy-particle therapy, **13-15, 13-16**
- High-voltage power tube, **6-72**
- Hold-up tank, **11-15**
- Hood, **9-5, 17-13, 22-5**
 - California, **9-6**
 - fume, **17-13**
 - handwork, **22-10**
 - lathe, **22-8**
 - process, **22-6, 22-8**
 - vacuum bench, **17-14**
- Hot, definition, **2-13**
- Hot cell, **17-23, 22-22**
- Housekeeping, **11-48**
- Human body (see Body; Standard man)
- Hydrogen, **8-4**
- Hygienic standard, **11-4**
- Identification of particle, **10-38**
- Image intensifier, **13-2**
- Immersed body, **14-9**
- Immobilization, **13-5**
- Impaction, **20-9**
- Impactor, **20-7**
 - annular, **20-8**
 - kinetic, **11-5**
- Impingement, **11-4**
- Impinger, **20-7**
- Incineration, **21-54, 22-4, 22-31, 22-42**
- Index of elements and atomic numbers, **1-3**
- Induction accelerator, betatron, **2-6**
- Industrial applications, **11-2 to 11-52**
- Industrial Hygiene Code, **3-34**
- Industrial medical department, **11-3**
- Inelastic scattering, **7-2, 7-11**
- Inertial collector, **22-32**
- Ingestion, **14-15**
- Inhalation, **11-3, 14-15**
- Installation, **3-40**
- Instruments, **10-1 to 10-84**
 - chemical, **10-76**
 - condenser, **10-10**
 - Geiger-Müller, **10-41**
 - glass, **10-79**
 - ionization, **10-3**
 - photographic, **10-53**
 - photoluminescent, **10-79**
 - pocket, **10-9**
- Instruments, proportional, **10-33**
 - scintillation, **10-14**
- Insulator, **10-7**
- Intake, rate, **11-10**
- Integral dose, **2-9**
- Intensifier, image, **13-6**
- Intensifying foil, **10-67, 10-68**
- Intensity, **2-13**
- Interaction process, **7-2**
 - neutrons, **7-11**
 - photons, **7-6**
- Interceptor probe, **12-24**
- Interior surface, **17-24**
- Interlock, **12-22, 13-17**
- Intermediate neutron, **7-11**
- Internal emitter, **13-24, 13-25**
- Internal organs, **19-12**
- Internal radiation, **11-3**
- Internal radioisotope, **13-23**
- Internal source, **13-19, 13-22**
- International Commission on Radiological Protection (ICRP), **3-4**
- International Commission on Radiological Units and Measurements (ICRU), **3-4**
- Interstate Commerce Commission (ICC), **3-43**
- Interstitial source, **13-18, 13-21, 13-24**
- Interstitial therapy, **13-22, 13-25**
- Intestine, lower large, **14-16**
- Intracavity source, **13-24**
- Intracavity therapy, **13-25**
- Intramural transfer, **17-32**
- Inverse-square law, **13-4**
- Iodine, **8-22**
- Iodine-131, **13-23**
- Ion, **2-13**
- Ion collection, **10-4**
- Ion exchange, **21-20, 21-22**
- Ion-exchange resins, **16-4, 21-20**
- Ionium, **16-21**
- Ionization, **2-13, 10-4**
 - specific, **2-21, 14-6**
- Ionization chamber, **10-3, 10-4**
 - charge or current measurement, **10-7 to 10-9**
 - design features, **10-6, 10-7**
 - extrapolation, **10-12, 10-13**
 - instruments, **10-9 to 10-11**
 - malfunction, causes of, **10-13**
 - standard free-air, **10-11, 10-12**
- Ionizing event, **10-37**
- Ionizing radiation, **5-2 to 5-14**
- Iron, **8-18, 8-33, 8-35**
- Irradiation, of fetus, **19-13**
 - of materials, **6-67**
- Isomorphous replacement, **21-14**
- Isotope, **2-14, 3-6, 11-45**
 - adsorption, **18-5**
 - handling, **11-46**
 - in hospitals, **21-9**
 - properties, **6-89 to 6-190**
 - radioactive, **6-89, 11-45**
- Isotopes Committee, **13-23**
- Isotopic purity, **15-8**

Junior cave, 17-22, 22-22

Kaolinite, 21-27

kc, 2-14

Keratosis, 19-9

kev, 2-14

Kilocurie (kc), 2-14

Klystron, 6-58, 6-72

Krypton, 10-48

Labeling, 3-25, 3-41

Laboratory, design, 9-1 to 9-11

medical radioisotope, 9-11

radiobiological, 9-10

radiochemical, 9-8

research, 12-3

Laboratory bench, 9-4

Laboratory furniture, 9-4

Laboratory hood, 11-48, 22-15

Laboratory plumbing, 9-7

Laboratory requirements, 16-2

Laboratory sink, 9-4

Laboratory surface, 9-3

Latent image, 10-58

Latent-image fading, 10-71

Lathe hood, 22-8

LD-50, 2-10, 19-3

Lead, 8-26, 8-34, 8-36

Lead shielding, 1-36

Leakage radiation, 8-46, 13-3

Leakage test, 11-40, 11-43

Legal (medical) problems, 23-14

Legislation, 3-17

Lens of the eye, 3-14

Leukemia, 19-15

Leukopenia, 19-3, 19-10

Life expectancy, 3-5

Life shortening, 19-7, 19-14

Life span, 19-7, 19-14

Linac, 6-43

Linear accelerator, 2-14, 6-43, 6-71

Liquid cleaning agent, 18-3

Liquid decontamination, 18-7

Liquid-drop contamination, 18-3

Liquid effluent, 9-2

Liquid hold-up, 18-5

Liquid reagent, 18-5

Liquid sampling, 20-29

Liquid-scintillation method, 16-8

Liquid scintillator, 10-19

Liquid waste, 21-8 to 21-12, 21-34 to 21-50

high-level, 11-16, 21-10, 21-35

Local exhaust, 22-6

Localizing device, 13-16

Logarithm tables (four-place), 1-8, 1-9

of exponentials, 1-26 to 1-29

Longevity, 19-14

Loose contamination, 18-14

Lymphopenia, 19-3

MAC, 11-4, 11-7

Magnesium, 8-10

Magnetic focusing, 6-59

Magnetron, 6-72

Major incidents, 18-21

Man (*see* Standard man)

Manipulation, 17-18

Manipulator, 17-7, 17-20, 17-29, 17-31

Manipulator cell, 17-24

Mass median diameter, 20-15

Mass stopping power, alpha radiation, 14-4

Master-slave manipulator, 17-29

Maximum allowable concentration (MAC), 11-4, 11-7

Maximum permissible body burden, 14-13, 14-14

Maximum permissible concentration (MPC), 3-8, 3-28, 14-9, 14-15

Maximum permissible dose (MPD), 2-9, 3-12, 3-19, 19-6, 19-7

Mc (megacurie), 2-15

mc (millicurie), 2-15

Mean free path, 2-14

Mean life, 2-15, 5-4

Measurement of radiation, 10-2 to 10-84

chemical method, 10-76

Geiger counter, 10-41

glass dosimetry, 10-79

ionization method, 10-2

photographic method, 10-53

proportional counter, 10-33

scintillation method, 10-14

Median lethal dose (LD-50), 2-10, 19-3

Medical control program, 23-6

Medical department, industrial, 11-3

Medical examination, 23-2

periodic, 23-6

Medical film, 13-3

Medical isotope, 13-19, 13-21

Medical patient, 13-3

Medical protection problems, 13-2

Medical radiation, 13-2 to 13-29

Medical records, 23-12

form, 23-13

retention, 23-13

Medicolegal problems, 23-14

Megacurie (Mc), 2-15

Mental and emotional factors, 23-4

Meson, 6-54

Metal, densities of, 1-35

heavy, 22-7

pyrophoric, 11-22, 22-7

Metal scrap, 11-24

Meteorite, 6-15

Meteorological control, 22-26

Meteorological factors, 21-8

Methane, 10-36

Mev, 2-15

Millicurie (mc), 2-15

Milling, 11-19, 22-2

Milling-machine enclosure, 22-10

Million electron volt (Mev), 2-15

Millipore filter, 20-6

Milliroentgen (mr), 2-15

Minimum filtration, 13-4

Mining, 22-2

Mirror system, 17-18

Mist, 20-2, 22-2

Mixed packing, label, transportation, 3-56

Mixed radiation, 10-39, 10-65

- Moderation, **15-3, 15-6**
 Moderator, **2-15, 6-74, 11-25**
 Molybdenum, **8-20**
 Monitor, **2-15**
 gamma, **12-23**
 neutron, **12-23**
 radiation, **12-18, 13-17**
 television, **12-24**
 Monitoring, **3-7, 10-65**
 area, **2-4, 2-15, 10-58**
 personnel, **2-15, 11-45, 12-4, 12-9**
 photographic, **10-53**
 Monte Carlo technique, **8-37**
 Montmorillonite clay, **21-27**
 MPC (*see* Maximum permissible concentration)
 MPD (*see* Maximum permissible dose)
 mr, **2-15**
 Multiple-unit arrays, **15-10**
 Multiplying factor, **15-7**
 Mutation, **19-16**

 n unit, **2-15**
 Nail, ridging, **19-9**
 Nalcite HCR, **21-21**
 Narrow-beam attenuation, **7-6**
 National Bureau of Standards (NBS) Handbooks, **3-2 to 3-30**
 National Committee on Radiation Protection and Measurements (NCRP), **3-2 to 3-30**
 Natural radiation dosage of the body, **4-17**
 Natural trigonometric functions, **1-10, 1-11**
 Neon, **10-48**
 Neutrino, **5-9**
 Neutron, **2-16, 7-10 to 7-13, 17-6**
 absorbed dose, **14-7**
 from accelerator, **6-50**
 biological effects, **13-15**
 delayed, **6-76**
 energy lost per collision, **14-7**
 fast, **7-11, 10-72, 14-6**
 giant-resonance, **6-53**
 high-energy, **6-53, 10-17**
 intermediate, **7-11**
 mixed, **10-73**
 relativistic, **7-11**
 resonance, **6-51, 7-11**
 thermal, **6-51, 7-10, 10-72, 14-7**
 Neutron absorbers, **15-9**
 Neutron beams, **12-17**
 Neutron capture, **7-12**
 Neutron counter, **10-40**
 Neutron energy distribution, **6-51**
 Neutron flux, fast, **6-51**
 Neutron leakage, **15-3**
 Neutron monitor, **12-23**
 Neutron monitoring, **10-70, 12-2**
 Neutron radiation, **3-11**
 Neutron yield, **6-53**
 Nitrogen, **8-7, 11-31**
 Noble gas, **22-44**
 Nonporous material, **18-5**
 Nuclear collision, **14-6**
 Nuclear data source, **6-10**
 Nuclear reactor (*see* Reactor)

 Nuclear safety, **15-2**
 Nuclear Standards Board, **3-32**
 Nucleon, **2-16**
 Nucleus, **2-16**
 Nuclide, **2-16**
 Number median diameter, **20-15**
 Numbers, squares, square roots, cubes and cube roots, **1-12 to 1-24**

 Occupancy factor, **3-14, 8-43, 8-45**
 Occupational disease, **11-2, 11-3**
 Occupational exposure, low-level, **19-9**
 Ocean, radioactivity of, **4-10**
 Ocean disposal, **3-10, 21-39 to 21-41**
 Off-site disposal cost, **21-54**
 Open trench, **21-41**
 Operating problems, **12-19**
 Operating-room X-ray, **13-7**
 Operator of nuclear reactor, **23-2**
 Optical methods (dosimetry), **10-79 to 10-84**
 Ore processing, **21-9, 22-2**
 Organ, critical, **3-14, 3-19, 14-13, 14-18, 19-12**
 target, **19-4**
 Organic scintillator, **10-18**
 Oscilloscope, **6-72**
 Overexposure, in emergencies, **23-10**
 reporting, **3-39**
 Oxygen, **8-8, 11-31**

 Paint, **9-3**
 Painted surface, **18-8**
 Pair production, **2-16, 6-48, 7-7, 10-16**
 Paper, glass, **20-6**
 Parent, definition, **2-16**
 Particle, **5-2**
 diameter, **20-17**
 identification, **10-38**
 Particle accelerator (*see* Accelerator)
 Particle size, **20-14, 20-18**
 biological effect, **20-15**
 Particulates, **22-2, 22-32, 22-43**
 in respiratory tract, **1-39**
 Particulate removal, **22-34**
 Pathological effects of radiation, **23-3**
 Patient, **13-3**
 care, **13-25**
 dose, **13-8, 13-25**
 Periscope, **17-27**
 over-the-wall, **17-18, 17-19**
 through-wall, **17-19**
 Permissible dose (*see* Maximum permissible dose)
 Personnel, decontamination, **18-19**
 exposure, **11-36**
 selection of, **23-2**
 Personnel control, **23-2 to 23-15**
 medical, **23-6**
 Personnel monitoring, **2-15, 11-45, 12-4, 12-9**
 Phase stability, **6-38**
 Phosphate coagulation, **21-19**
 Phosphor, **10-15, 11-49**
 anthracene, **10-19**
 liquid, **10-19**
 organic, **10-17, 10-18**
 plastic, **10-20**

- Phosphor, sodium iodide, **10-18**
 stilbene, **10-19**
 Phosphorus, **8-13**
 Phosphorus-32, **13-23**
 Photoelectric absorption, **10-16**
 Photoelectric effect, **7-6**
 Photo-fluorograph, **13-7**
 Photographic monitoring, **10-53 to 10-75**
 Photomultiplier, **10-21, 10-24**
 commercial, **10-25**
 Photon, **2-16**
 Photon interaction, **7-5**
 Photoneutron, **6-53**
 Photonuclear process, **6-52**
 Physical examination, **11-7, 11-12, 11-45**
 Physical requirements, **23-5**
 Physiological effects of radiation, **19-2 to 19-17**
 Fig. **17-11, 17-32**
 Pipe intersection, **15-10**
 Pitchblende, **11-19**
 Planck's constant, **2-16**
 Plant liquid effluent, **11-14**
 Plastic phosphor, **10-20**
 Plateau, **10-44**
 Platinum, **8-24**
 Plumbing, **9-7**
 Plutonium, **22-4**
 analysis, **16-16**
 determination, **16-16, 16-17**
 recovery, **21-11**
 Pocket chamber, **2-16**
 Pocket dosimeter, **2-16, 10-9**
 Poisonous material, **11-2**
 Polarity, **10-6**
 Pollution, air, **11-12**
 stream, **11-14, 21-2**
 water, **11-14**
 Polonium, **11-18**
 determination, **16-22**
 in solution, **16-22**
 Polyatomic gas, **10-36**
 Porous material, **18-7**
 Positron, **2-17**
 Potassium, **6-31, 8-16**
 in living subjects, **4-15**
 minerals, **6-15**
 and rubidium in rock, **4-9**
 Power supply, **10-26**
 Precipitator, chemical, **21-13**
 electrostatic, **11-4, 20-6, 22-34**
 thermal, **20-10**
 Predissociation, **10-44**
 Prefilter, **22-37**
 Primary barrier, **13-4**
 Probe, interceptor, **12-24**
 Process cell, **17-24**
 Process hood, **22-6, 22-8**
 Production facility, definition, **2-17**
 Proportional counter, **10-33 to 10-41**
 Proportional region, **10-38**
 Protactinium analysis, **16-9**
 Protection, history, **3-3**
 regulations, **3-2 to 3-35**
 standards, **3-35 to 3-72**
 Protective agents, **19-4**
 Protective clothing, **11-23, 11-49, 12-8**
 Protective coating, **18-10**
 Protective tube, **13-11**
 housing, **6-63**
 Proton, **2-17, 6-46**
 Protraction, **19-2**
 Pulse height, **10-22**
 Purification of water, **21-6**
 Pyrophoric metal, **11-22, 22-7**
 Qualified expert, definition, **3-19**
 Quality, **13-11**
 Quench, **10-44**
 Quenching, **2-17, 10-45**
 Quenching circuits, **10-45**
 r, **2-17**
 r meter, **2-19**
 Rad, **2-17, 3-19, 14-2**
 Radarscope, **6-72**
 Radiation, **2-17, 10-38**
 from accelerator, **6-45 to 6-60**
 annihilation, **2-17**
 background, **2-17**
 beta, **7-2**
 characteristic, **2-17**
 detection and measurement, **10-2 to 10-84**
 from electrical sources, **6-61 to 6-73**
 external, **11-10, 11-21, 14-2**
 gamma, **7-5**
 interaction with matter, **7-2 to 7-13**
 ionizing, **5-2**
 leakage, **8-46, 13-3**
 neutron, **7-10**
 90° scatter, **8-48**
 scattered, **8-47, 11-36, 13-3**
 X-ray, **7-5**
 Radiation detection, **10-2 to 10-84**
 (See also Measurement of radiation)
 Radiation effect, immediate, **19-8**
 late, **19-9**
 Radiation equilibrium, **10-6**
 Radiation exposure, control, **17-2**
 previous, **23-3**
 Radiation hazard, color, **3-32**
 external, **11-47**
 Radiation injury, **23-14**
 Radiation length, **6-49**
 Radiation monitor, **12-18, 13-17**
 Radiation protection (see Protection; Shielding)
 Radiation quality, **10-65, 10-67**
 Radiation regulations, **3-2 to 3-35**
 Radiation safety, **12-2**
 Radiation safety supervisor, **3-41**
 Radiation source, **6-1 to 6-190**
 control, **22-5**
 electrical, **6-61 to 6-73**
 gamma ray, **11-37**
 handling, **11-33**
 material, **2-20, 6-75**
 natural, **6-2 to 6-32**
 nuclear reactor, **6-74 to 6-88**
 particle accelerator, **6-33 to 6-60**

- Radiation source, preparation, **13-22**
 - radioactive isotope, **6-89 to 6-190**
 - sealed, **3-19, 11-37**
- Radiation survey, **13-10, 13-18, 13-27**
- Radiation therapy (*see* Radioactive material; Teletherapy; X-ray therapy)
- Radioactive contaminants, **22-2**
- Radioactive decay, **5-6**
- Radioactive dust, **14-15**
- Radioactive elements, half-lives, **1-4**
 - in rock, **4-9**
- Radioactive gas, **22-7, 22-25**
 - inert, **11-8**
- Radioactive isotope (*see* Isotope)
- Radioactive material, leakage, **11-43**
 - in medicine (diagnosis and therapy), **13-18**
 - natural, **6-15, 11-16**
 - properties, **17-5**
 - shipment, **11-24**
 - spilled, **11-41**
- Radioactive source, handling, **11-41**
 - storage, **11-41**
- Radioactivity, of air, **4-12**
 - artificial, **2-4**
 - definition, **2-18**
 - of earth's crust, **4-7**
 - of human body, **4-15**
 - induced, **6-54**
 - natural, **6-2 to 6-32**
 - of ocean, **4-10**
 - of rock, **6-15**
 - of water, **4-10, 4-11**
- Radioautograph, **2-18**
- Radiobiology, **2-18**
- Radiographic installation, **13-5**
- Radiography, industrial, **11-35**
 - medical, **6-63, 6-66, 13-4**
- Radioisotope (*see* Isotope)
- Radionuclide, **5-2, 5-4**
 - gamma radiation, **6-10**
 - natural, **6-2 to 6-15**
- Radium, **3-9, 6-11, 11-38, 17-6**
 - in body, **4-16**
 - coprecipitation, **16-13**
 - counting, **16-12**
 - determination, **16-12**
 - gamma radiation, **6-13**
 - isolation, **16-12**
 - in ocean, **4-9**
 - in rock, **4-9**
- Radium-226, **6-11**
- Radium B (Pb-214), **6-11**
- Radium C (Bi-214), **6-12**
- Radium C'' (Tl-210), **6-12**
- Radium capsule, **11-39**
 - rupture, **11-39**
- Radium content, of rock, **4-8**
 - ocean, **4-9**
 - of water, **4-11, 4-12**
- Radium dial painting, **11-49**
- Radon, **4-12, 6-11, 11-17**
 - concentration in mines, **4-14**
 - decay products, **20-26**
- Range, of particles in air, **12-12**
 - of protons, **6-46**
- Rate of settling, **20-18**
- Rayleigh scattering, **7-7**
- RBE (*see* Relative biological effectiveness)
- RBE dose, **3-14**
- Reactivity, **6-76**
- Reactor, accidents, **11-34, 12-16**
 - air-cooled, **22-5, 22-44**
 - aqueous homogeneous, **6-79**
 - boiling-water, **6-79**
 - classification, **11-25**
 - converter, **6-80**
 - experimental, **11-33**
 - fast, **6-78, 11-25**
 - fused salts, **6-79**
 - gas, **6-79**
 - heterogeneous, **6-79**
 - boiling-water, **6-82**
 - fast-breeder, **6-83**
 - gas-cooled, **6-84**
 - pressurized-water, **6-80**
 - research, **6-86, 11-26**
 - homogeneous, **6-78, 11-26**
 - intermediate, **6-78, 11-25**
 - liquid-metal-fuel, **6-79, 6-80**
 - organic moderated and cooled, **6-79, 6-83**
 - pressurized-water, **6-79**
 - research, **12-16**
 - runaway, **18-21**
 - safety features in design, **12-17**
 - site of, **11-34, 21-3**
 - sodium-cooled graphite, **6-83**
 - start-up, **12-17**
 - thermal, **6-78, 11-25**
 - research, **6-85**
- Reactor canal, **11-33**
- Reactor coolant, **6-79, 11-29**
 - liquid metal, **11-30**
 - water, **11-29**
- Reactor-core manufacturing, **11-20 to 11-25**
- Reactor fuels, **6-78, 11-25**
 - processing, **21-10, 21-11**
- Reactor materials, **11-25**
- Reactor operation, **11-25**
- Reactor operator, **23-2**
- Reactor operator's license, **3-37**
- Recombination, **2-18, 10-5**
- Recovery time, **2-18, 10-36, 10-44, 10-47**
- Rectifier tube, **6-72**
- Reference data, **1-43**
- Registration, **3-41**
- Regulation, **3-31 to 3-72, 11-14**
- Reignition, **10-44**
- Relative biological effectiveness (RBE), **2-18, 2-19, 3-19, 14-9, 19-2**
 - alpha particles, **14-13**
 - fast neutrons, **14-13**
- Relative hazard, **14-14**
- Relativistic neutron, **7-11**
- Release, **11-13**
- Rem, **2-19, 14-2**
- Remote control, **17-10**
- Remote-control equipment, **17-18**
- Rep, **2-19, 14-2**
- Reporting overexposure, **3-39**
- Research applications, **12-2 to 12-26**

- Research laboratory, **12-3**
- Research reactor, **12-16**
- Resolution, **10-14**
- Resolving time, **10-36, 10-45, 10-46**
- Resonance accelerator, **6-34**
- Resonance generator, **13-13**
- Resonance neutrons, **6-51, 7-11**
- Resonance peak, **7-11**
- Resonant transformer, **6-69**
- Respiratory protection, **12-8**
- Retention tank, **21-21**
- Rise time, **10-36**
- Rock, igneous, **6-31**
 - radioactive elements in, **4-9**
 - radioactivity of, **6-15**
 - radium content of, **4-8**
 - sedimentary, **6-31**
- Roentgen, **2-19**
- Roentgen equivalent, man, **2-19, 14-2**
 - physical, **2-19, 14-2**
- Rotameter, **20-22**
- Route of entry, **11-3, 11-7**
- Runaway reactors, **18-21**
- Rupture, **22-5**
- Ruthenium-106, determination, **16-24**
- Safety, accelerator, **12-20**
 - nuclear, **15-2**
 - research reactor, **12-16**
- Safety factors, **15-4**
- Safety rod, **2-19**
- Salt dome, **21-44**
- Sampler-counter, **20-25**
- Sampling, **20-30**
 - air, **11-4**
 - biological, **11-9**
 - continuous, **11-6**
 - gas, **20-12, 20-14**
 - grab, **20-30**
 - isokinetic, **20-28**
 - liquid, **20-29**
 - stack, **20-27**
- Sampling equipment, **20-2 to 20-31**
- Sampling head, **20-22**
- Sampling pump, **20-21**
- Sampling techniques, **20-25**
- Sarcoma, **19-15**
- Saturation, **10-5**
- Saturation current, **2-19**
- Saturation voltage, **2-19**
- Scaler, **2-19**
- Scattered radiation, **8-47, 11-36, 13-3**
- Scattering, **2-19**
 - elastic, **7-2, 7-11**
 - inelastic, **7-2, 7-11**
 - Rayleigh, **7-7**
- Scavenger precipitate, **21-14**
- Scintillation, **2-20, 10-14**
- Scintillation counter, **10-14 to 10-32**
 - characteristics, **10-20**
 - organic, **10-18**
- Sea disposal of waste, **21-39**
- Seal coat, **18-9**
- Sealed sources, **3-19, 11-37**
- Secondary barrier, **8-53, 13-4**
- Secular equilibrium, **2-11**
- Sedimentation, **11-5**
- Sedimentation chamber, **20-11**
- Self-absorption, **2-20, 20-26**
- Self-luminous material, **11-49**
 - disposal, **11-50**
 - repair, **11-50**
- Self-quenching counter tube, **10-45**
- Sensitivity of film, **10-56, 10-62, 10-70**
- Septicemia, **19-3**
- Settling chamber, **20-11**
- Settling rate, **20-18**
- Settling velocity, **20-17**
- Shale formation, **21-44**
- Shield, **18-12**
 - clear plastic, **17-12**
 - close, **9-4**
 - (See also Shielding)
- Shielded syringe, **17-12**
- Shielding, **9-2, 12-6, 17-7**
 - accelerator, **6-54**
 - linear, **6-58**
 - betatron, **6-57**
 - brick, **1-37**
 - cesium-137, **8-68, 8-69**
 - cobalt-60, **8-60, 8-66**
 - concrete, **1-36, 1-37**
 - cyclotron, **6-57**
 - dental, **8-55**
 - electrostatic generator, **6-56**
 - laboratory, **9-4**
 - lead, **1-36**
 - portable, **9-4**
 - for fume hood, **17-22**
 - radiographic, **8-54**
 - synchrocyclotron, **6-58**
 - synchrotron, **6-58**
 - proton, **6-59**
 - therapeutic, **100-kv, 8-56**
 - 150-kv, **8-57**
 - 200-kv, **8-58**
 - 250-kv, **8-59**
 - voltage multiplier, **6-56**
 - X-ray, industrial, **8-60, 8-61**
 - 1-million-volt, **8-62**
 - 2-million-volt, **8-63**
- Shielding barrier (see Barrier)
- Shielding material, **8-41, 11-38**
 - gamma, **17-25**
 - low-atomic-number, **8-41**
 - transparent, **17-25**
- Shielding window, **17-24**
- Shipping of radioactive material, **11-24**
- Shipping container, **17-33**
- Shoe-fitting fluoroscope, **3-34, 13-7**
- Silicon, **8-12, 18-10**
- Site selection, **21-3**
- Site survey, **12-4**
- Skin cancer, **19-15**
- Skin damage, **19-7**
- Skin dose, **2-10**
- Skin reaction, **19-8**
- Smog, **20-2**
- Smoke, **20-2, 22-2**
- Sodium, **8-9**

- Sodium iodide, **8-29, 10-18**
- Solid waste, **21-50 to 21-60**
- Solvent extraction, **16-5**
- Source (*see* Radiation source)
- Source material, **2-20, 6-75**
- Special nuclear material, **2-20, 6-75**
- Specific activity, **2-21**
- Specific ionization, **2-21, 14-6**
- Spectrometry, **10-22**
- Spill index, **18-8, 18-10**
- Spills, minor, **18-20**
- Spleen transplant, **19-4**
- Spurious count, **10-45**
- Stack dispersion, **22-26**
- Stack-effluent monitoring, **22-45**
- Stack factor, **22-27**
- Standard free-air chamber, **10-11**
- Standard man, **1-38, 1-39, 14-10**
- Stang precipitator, **17-11**
- State code, **11-2**
- State regulation, **3-39**
- Static-elimination unit, **3-37**
- Stepped wedge, **10-66**
- Stever diagram, **10-47**
- Stilbene, **10-19**
- Stokes' law, **20-17**
- Stopping power, **2-21**
 - alpha radiation, **14-4**
 - beta radiation, **14-5**
- Storage, **11-41, 13-26, 17-32**
 - liquid waste, **21-35**
- Storage equipment, **17-32**
- Storage well, **17-32**
- Straggling, **7-3**
- Stratification, **20-30**
- Stream pollution, **11-14, 21-2**
- Strip coat, **18-10**
- Strong magnetic focusing, **6-59**
- Strontium, **21-13**
 - removal, **21-19**
- Strontium-90, **16-5**
 - determination, **16-7**
 - in urine, **16-5**
 - in waste, **21-50**
- Sulfur, **8-14**
- Superficial oxide layer, **18-8**
- Superficial therapy, **13-12**
- Supervoltage therapy, **13-13**
- Surface, interior, **17-24**
- Surface adsorption, **21-14**
- Surface coat, **18-9**
 - permanent, **18-9**
- Surface contamination, **11-21, 12-9, 18-2 to 18-23**
 - measurement and evaluation, **18-12**
- Surface source, **13-18**
- Survey, radiation, **13-10, 13-18, 13-27**
- Survey instrument, **2-21, 10-24**
- Survey meter, **10-11**
- Swimming pool, **17-31**
- Symbols and signs, **1-6**
- Synchrocyotron, **2-21, 6-40**
- Synchrotron, **2-21**
 - alternating-gradient, **6-59**
- Synchrotron, electron, **6-39, 6-59**
 - proton, **6-42**
- Tank, hold-up, **11-15**
- Target organ, **19-4**
- Telangiectasia, **19-9**
- Teletherapy, **13-18, 13-19**
- Television monitor, **12-24**
- Tell manipulator, **17-31**
- Thallium, **8-25**
- Therapy (*see* Radioactive material; Teletherapy; X-ray therapy)
- Thermal neutron, **6-51, 7-10, 10-72, 14-7**
 - absorbed dose, **14-7**
- Thermal precipitation, **11-5**
- Thermal precipitator, **20-10**
- Thermionic emission, **10-23**
- Thick target, **6-49, 6-51**
- Thin target, **6-49**
- Thoriated electrode, **11-50, 11-51**
- Thorium, **11-50, 16-21, 22-4**
 - natural, **16-19**
 - in rock, **4-8**
 - with thorin, measurement, **16-20**
- Thorium-232 series, **6-9**
- Thorium metal, **22-4**
- Thorium minerals, **6-15**
- Thorium refinery, **11-51**
- Thorium series, **6-4**
- Thoron, **4-12**
- Thoron decay products, **20-26**
- Thorotrast, **13-5**
- Threshold, **10-43**
- Threshold dose, **2-10**
- Threshold limit, **3-34**
- Thrombocytopenia, **19-3**
- Time lag, **10-49**
- Time limitation, **12-5**
- Time resolution, **10-23**
- Timing, **13-17**
- Tin, **8-21, 8-34, 8-35**
- Tissue dose, **2-10**
- Tissue-equivalent material, **2-21**
- Tolerance dose, **2-10**
- Tongs, **17-7, 17-8, 17-20**
- Topography, **21-7**
- Total body radiation, **19-2**
- Total contamination, **18-14**
- Townsend coefficient, **10-35**
- Toxic materials, **11-2**
- Toxicity, **12-3**
- Track counting, **10-71, 10-73**
- Transient equilibrium, **2-11**
- Transition, **5-11**
- Transportation, **3-42 to 3-72, 17-32 to 17-35**
 - by motor vehicle, **3-61**
- Transportation placard, **3-59, 3-60**
- Treatment, excessive dose, **23-11**
- Trench, open, **21-41**
- Trigonometric functions, natural, **1-10, 1-11**
- Tritium analysis, **16-7**
- Tritium oxide, conversion, **16-7**
- Tube, gas-filled, **10-44**
 - Geiger-Müller, **10-41**
 - high-voltage power, **6-72**

- Tube, klystron, 6-58, 6-72
 - radium, 11-34
 - rectifier, 6-72
 - X-ray (see X-ray tube)
- Tungsten, 8-23, 8-34
- Underwater facilities, 17-6, 17-31
- Underwater storage, 17-32
- U.S. Coast Guard, 3-66
- U.S. Postal Regulations, 3-70
- Units commonly used, 1-7
- Uranium, 6-31, 8-27, 8-36
 - analysis, 16-10
 - in blood, 16-15
 - chips, turnings, and fines, 11-23
 - dust, 11-20
 - enriched, 11-21
 - extraction, 16-11
 - fire, extinguishing, 11-22
 - fluorimetric analysis, 16-14
 - fumes, 11-20
 - isolation, 16-10
 - machining, 11-23
 - mining and milling, 11-17
 - natural, 11-16, 11-21
 - partially enriched, 15-9
 - in rock, 4-8
- Uranium-235 series, 6-8
- Uranium-238 series, 6-6
- Uranium hexafluoride (UF₆), 22-4
- Uranium metal, 22-4
- Uranium minerals, 6-15
- Uranium-zirconium alloy, 11-22
- Urine, routine examination, 16-22
 - sample, 11-9
- Use factor, 8-43
- Useful beam, 2-21
- Utilization facility, definition, 2-21
- Vacuum-bench enclosure, 17-14
- Vacuum cleaning, 18-7
- Van de Graaff generator, 6-34, 13-13
 - electrostatic, 6-70
- Ventilating, 9-7
 - general, 22-23, 22-24
 - laboratory, 9-7
- Viewing device, 17-24
- Viewing equipment, 17-7
- Viewpoint, 13-17
- Volatilization, 16-4
- Voltage multiplier, 6-33
- Volume dose, 2-9
- Wall surface, 18-10
- Warning device, 13-7
- Waste, clay-bearing bed, 21-45
 - decontamination, 21-9
 - high-level liquid, 11-16, 21-10, 21-35
 - high-level radioactive, 21-22
 - high-volume low-level, 21-14
 - laboratory, 21-9
 - liquid, 21-8
 - low-level, 21-8
 - solid, 21-50
- Waste disposal, 3-26, 3-27, 21-1 to 21-63
 - adsorption on clay, 21-23 to 21-27
 - biological processes, 21-33, 21-34
 - carbon-14, 3-9
 - chemical precipitation, 21-13 to 21-20
 - evaporation, 21-27 to 21-33
 - industrial, 11-24, 11-25, 11-32
 - iodine-131, 3-7
 - liquid, 21-8 to 21-12, 21-34 to 21-50
 - medical, 13-27
 - miscellaneous methods, 21-49
 - in ocean, 3-10, 21-39 to 21-41
 - open pit, 21-44
 - philosophy, 21-2, 21-3
 - phosphorus, 3-2, 3-7
 - regulations, 3-27
 - site selection, 21-3 to 21-8
 - solid, 21-50 to 21-60
 - standards, 3-7, 3-9, 3-10
- Water, 8-28, 8-33, 8-35
 - contaminated, 22-45
 - ground, 21-7
 - radioactivity of, 4-10, 4-11
 - radium content of, 4-11, 4-12
- Water boiler, 6-87
- Water purification, natural, 21-6
- Water supplies, 21-4
- Wax, 9-3, 18-10
- Weather data, 1-31 to 1-34
- Wet handling, 22-6
- Wigner effect, 2-22
- Window, 10-49
 - composite, 17-27
 - glass, 17-26
 - liquid, 17-26
- Wiping technique, 18-14
- Work habits, 11-48
- Workload, 3-14, 8-43
- Workmen's compensation law, 11-2, 23-14
- World Health Organization (WHO), 3-4
- Xenon-133, 14-9
- X-ray, 2-22, 6-61, 7-5, 13-3
 - from accelerators, 6-48
 - anticrime, 6-67
 - attenuation, 8-37
 - characteristic, 7-5
 - continuous, 7-5
 - dental, 6-63
 - diagnostic, 13-4
 - diagnostic exposure, 23-3
 - diffraction, 6-67
 - industrial, 11-35
 - medical, 6-63, 13-4
 - operating-room, 13-7
 - procedures, 23-7
 - shielding, 11-35
 - shoe fitting, 3-34, 6-67
 - thickness gauging, 6-67
 - wavelength, 6-61
- X-ray control, 6-68
- X-ray diagnosis, 13-4 to 13-11, 23-3, 23-7
- X-ray generator, 6-68, 6-69
- X-ray image-converter, 6-72

X-ray protection, 3-6, 3-10
X-ray spectrum, 6-61
X-ray therapy, 13-11 to 13-18
 contact, 13-12
 deep, 13-13
 high-energy particle, 13-15
 multimillion-volt, 13-14
 rotational, 13-17

INDEX

X-ray therapy, superficial, 13-12
 supervoltage, 13-13
X-ray tube, 6-61, 6-62, 6-64 to 6-66
 air-cooled, 6-63
 head, 13-4

Zeolite, 21-20
Zirconium, 22-4

14 DAY USE

RETURN TO DESK FROM WHICH BORROWED

PUBLIC HEALTH LIBRARY

This book is due on the last date stamped below, or on the date to which renewed.

Renewed books are subject to immediate recall.

APR 26 1967	DEC 26 1972
APR 22 1967	MAR 20 1973
SEP 11 1967	DEC 7 1976
11 OCT '67	NOV 29 1976
13 NOV. 67	DEC 03 1982
13 DEC 67	Renewed 11/26
13 JAN 68	Due end of winter quarter
JAN 15 1968	REC. PUBL. FEB 07 '83
NOV 14 1969	
NOV 25 1969	
NOV 25 1969	
OCT 2 1970	
OCT 6 1970	
Due end of SPRING Quarter subject to recall after	APR 26 1972
JUN 14 1972	

LD 21-40m-10,'65
(F7763s10)476

General Library
University of California
Berkeley

PUBLIC HEALTH LIBRARY



05885